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THORIUM BREEDER REACTOR EVALUATION.
PART I. FUEL YIELD AND FUEL CYCLE
COSTS IN FIVE THERMAL BREEDERS

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REACTOR DIVISION

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ABSTRACT

The performances of aqueous-homogeneous (AHBR), molten-salt (MSBR), liquid-bismuth (LBBR), gas-cooled graphite-moderated (GGBR), and deuterium-moderated gas-cooled (DGBR) breeder reactors were evaluated in respect to fuel yield, fuel cycle costs, and development status. A net electrical plant capability of 1000 Mwe was selected, and the fuel and fertile streams were processed continuously on-site.

The maximum annual fuel yields were 16, 7, 4, 4, and 4.5 %/yr respectively at a fuel cycle cost of 1.5 mills/kwhr. The minimum estimated fuel cycle costs were 0.9, 0.6, 1.0, 1.2, and 1.3 mills/kwhr at fuel yields of 7, 1, 1, 2, and 3 %/yr. At a fuel yield of 4%/yr, the costs were 0.9, 0.9, 1.5, 1.5, and 1.3 mills/kwhr. Only the AHBR and the MSBR are capable of achieving fuel yields substantially in excess of 4%/yr, and therefore, in view of the uncertainties in nuclear data and efficiencies of processing methods, only these two can be listed with confidence as being able to satisfy the main criterion of the AEC long-range thorium breeder program, viz. a doubling time of 25 years or less.

The development effort required to bring the various concepts to the stage where a prototype station could be designed was estimated to be least for the AHBR, somewhat more for the MSBR, and several times as much for the other systems.

The AHBR was judged to rank first in regard to nuclear capability, fuel cycle potential, and status of development.

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* Appendices are issued under separate cover using same title as this
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1. 1. 1.

1. S U M M A R Y

As part of the ORNL responsibility for guiding the AEC Thermal Thorium Breeder Program, an evaluation of types of reactors capable of efficient utilization of thorium was initiated at the Laboratory in July 1959. This report presents the results of the evaluation to date which has been primarily devoted to an investigation of the relative fuel cycle potentials of the most promising thorium breeder reactors.

An initial survey of a wide variety of reactor types indicated that five types in particular warranted detailed consideration. These are the aqueous homogeneous (AHBR), molten salt (MSBR), liquid bismuth (LBBR), gas-cooled graphite (GGBR), and deuterium-moderated gas-cooled (DGBR). As a basis for evaluating the merits of these systems, a conceptual design was developed for each which was aimed at optimizing both fuel cycle costs and fuel yield (reciprocal of doubling time). These two criteria were selected as figures of merit since they measure, respectively, the short term as well as the long term potential of each system for achieving economic power with the thorium cycle.

To put the evaluation on a uniform basis, all plants were assumed to have the same total electrical capacity-1000 electrical megawatts (Mwe) net; however, the total thermal output and the number of reactors per station varied, depending on the characteristics of each type of reactor and associated steam plant. The reactor design and operating conditions selected for the study, however, are believed to represent the maximum thermal performance that can be achieved with existing technology.

For uniformity, continuous on-site processing of fuel and fertile material was assumed in all cases, with the size of the processing plants and the method of processing varying from system to system. An attempt was made to develop processing costs which accurately reflect the type of process and the amounts and nature of the material being handled since the processes considered are in varying stages of development, the processing cost picture involves a considerable margin of uncertainty and further refinement is desirable.

The aqueous homogeneous reactor design selected for evaluation utilizes a cylindrical re-entrant type Zircaloy-2 core, 12-15 ft long x 4-5 ft in diameter, surrounded by a blanket region 2 ft thick on the sides. This blanket region consists of three equal-volume annuli of thorium oxide pellets, retained between perforated cylinders of Zr-2. The pellet regions are further divided circumferentially into 33 sectors, each containing a separate batch of pellets. To minimize protactinium losses, batches of pellets in high flux regions are interchanged daily with batches in low flux regions by hydraulic transport with D_2O . After a period of from 100 to 600 days each batch of pellets is removed from the reactor and sent to a Thorex processing plant. In the proposed design, the fuel solution leaves the reactor at $554^{\circ}F$, producing steam at $500^{\circ}F$ and 416 psia. The thermal output of each reactor is 910 Mw, and the net electrical output is 250 Mwe; thus four reactors are required for the station. Details of the bases used in arriving at this concept are given in the text.

The molten salt reactor concept has a cylindrical core, ~ 8 ft in diameter and ~ 8 ft long, surrounded by a 3-ft blanket on all sides. A graphite reflector 1 ft thick surrounds the blanket. The fuel stream heat exchanger and fuel pump are mounted directly above the reactor in a vertical configuration to minimize fuel inventory and facilitate maintenance of these components. The fuel salt flows in some 90 two-pass bayonet tubes of impermeable graphite, which are inserted in vertical passages in the graphite moderator structure. The region between the core and the reactor vessel, and the annulus between the fuel tubes and moderator, are filled with blanket fluid. Volatile fission products are removed from the fuel salt by helium sparging and nonvolatile fission products by a combination fluoride volatility process and HF dissolution. Since the molten salt leaves the reactor at $1300^{\circ}F$, steam conditions ($1050^{\circ}F$, 1800 psi) are comparable to those of a modern coal-fired plant resulting in a net station efficiency of 42.3%. Each of two molten salt reactors produces 1182 Mw of heat in the 1000 Mwe plant.

The liquid bismuth breeder concept chosen for evaluation has characteristics similar to the MSBR; however, in this case the graphite-moderated core is 6-1/2 ft in diameter x 6-1/2 ft long, surrounded by a 4 ft thick blanket (50 vol % graphite) and 1 ft graphite reflector. Both fuel and fertile stream heat exchanger and pumps are located inside the 12 ft diameter

x 34 ft long reactor vessel to minimize inventories. Volatile fission products are removed by helium sparging and nonvolatiles partly by an extraction process using chlorides of potassium, sodium, and magnesium and partly with a crystallization process and zinc slagging. The steam conditions are the same as the MSBR, resulting in a net station efficiency of 42.1%. Each of the two reactors in the station produces 1188 Mw of heat.

The gas-cooled graphite reactor consists of a horizontal cylindrical core 4-5 ft in diameter and 20 ft long, surrounded by a 2-ft thick blanket plus a 1 ft graphite reflector and 1 ft iron thermal shield. The core region consists of intersecting planes of permanent structural graphite about 1 in. thick forming square fuel channels 4-6 in. I.D. The fuel elements, consisting of unclad graphite plates impregnated with UO_2 , are loaded during operation from each end of the core. The blanket consists of concentric rings of ThO_2 pellets which are mixed and processed in a manner similar to that employed for the AHBR. The helium coolant leaves the reactor at 1500°F producing steam at 1050°F and 1450 psi and yielding a net station efficiency of 36.1%. Each of four reactors in the station has a thermal capacity of 692 Mw.

The deuterium-moderated gas-cooled reactor (DGBR) has a horizontal cylindrical core 7.27 ft in diameter x 20 ft long, surrounded by a 1.0-ft blanket and 1-ft iron thermal shield. The fuel elements consisting of two concentric unclad graphite cylinders, impregnated with UO_2 , are cooled by helium which enters the reactor at 525°F and 2000 psia and leaves at 1500°F . These fuel tubes are enclosed in a 1/4-in.-thick graphite casing which is inside a second graphite casing coated with polished zirconium and surrounded on both sides with stagnant gas to prevent heat loss to the colder D_2O moderator. In order to improve the breeding ratio of the reactor, the core also contains some ThO_2 in the form of pellets inside perforated Zr-2 tubes. These, as well as the blanket pellets, are cooled by D_2O . The blanket is similar to that of the AHBR, consisting of three concentric rings, each divided into 20 sectors. Batches of pellets from each sector are mixed daily and reprocessed as previously described for the AHBR. Steam at 1050°F , 1450 psi is produced, yielding a typical net station efficiency of $\sim 32\%$. Each of four reactors in the station has a thermal capacity of 775 Mw.

Using the aforementioned concepts as a basis, nuclear calculations were performed by means of the 34-group, one-dimensional, multiregion GNU program for the IBM-704. This program provides a "criticality search" by means of which either a dimension or one or more concentrations are varied until the multiplication constant is unity, at which point the fractions of neutrons involved in capture of fission reaction with each species is established relative to the mean density of that species in each region. These data are then used in the equilibrium reactor calculation program for the IBM-704 along with specifications such as reactor power, fuel volume, fertile stream volume, processing rates, etc. to compute the equilibrium concentrations of protactinium, uranium isotopes and fission products. The constraints imposed on the system of equilibrium equations are the conservation of mass, criticality, and neutrons. Finally, the nuclear data were used in conjunction with established cost bases to determine fuel cycle costs.

The results of the above calculations are presented graphically in Fig. 1.1, which shows the relation between total fuel cycle costs and fuel yield (% annual fuel yield = $100 \div$ doubling time in years). As seen in this figure, only the AHBR and the MSBR are capable of achieving fuel yields substantially in excess of 4%/yr, and therefore, in view of the uncertainties in nuclear data and efficiencies of processing methods, only these two can be listed with confidence as being able to satisfy the main criterion of the AEC long-range thorium breeder program.

Reactors with the highest fuel yield have fuel costs which are least adversely affected by rising costs of fissionable material. From this standpoint, the aqueous homogeneous type of breeder appears to be the best of the systems considered.

Fig. 1.1 also shows that, except for the AHBR, minimum fuel cycle costs occur at fuel yields of the order of 2%/yr, corresponding to doubling times of about 50 years. These fuel cycle costs are based on present AEC fuel inventory charges and fissionable material base charges, and as long as these charges prevail, all of the proposed systems, except the AHBR,

Fig. 1.1. Thermal Breeder Evaluation.

would be operated as autonomous (self-fueled) reactors, and the excess fuel produced would have little importance in expanding the nuclear economy. Judged in respect to minimum fuel cycle cost, the MSBR appears to be the most attractive system, based on current prices.

The integrity of the core vessel and handling of the fertile stream in the AHBR remain to be demonstrated; cost estimates are reasonably reliable. In the MSBR, the integrity of the core structure and the effectiveness of xenon removal are critical unknowns; estimates of processing costs are relatively less firm than in the AHBR. The same applies to the LBBR, with the addition that handling of a slurried fertile stream is more complicated and the fuel cycle costs are more nebulous. On-stream fueling is a major problem in both the GGBR and DGBR, and the core structure of the latter is complicated and presents hazards; fuel processing costs for both are rough estimates that probably err on the side of optimism.

Another significant aspect of the results presented in Fig. 1.1 is that three of the five concepts are predicted to be capable of fuel cycle costs below 1.5 mills/kwhr, while the AHBR and the MSBR show fuel costs less than 1 mill/kwhr. Therefore, if the costs of building and maintaining the radioactively contaminated cooling systems associated with these two concepts can be kept within reasonable limits, all five thorium breeder systems have a chance of meeting the intermediate range AEC goal of nuclear power at 6.0 mills/kwhr or less, with the probability of success decreasing in the following order: AHBR, MSBR, GGBR, DGBR, LBBR.

2. I N T R O D U C T I O N

Following the recommendations of an Ad Hoc Advisory Committee on Reactor Policies and Programs (Report to Commission, January 2, 1959), the U. S. AEC adopted as one of the objectives of the civilian nuclear power program "to develop breeder type reactors to make full use of the nuclear energy latent in both uranium and thorium" (John A. McCone, Chairman, U. S. AEC - April 7, 1959, speech at Edison Electric Institute, entitled "Government, Industry, and Nuclear Power"). To implement this objective relative to the utilization of thorium, the AEC assigned the Oak Ridge Operations Office the responsibility for a long range program directed to the development of a thermal (thorium) breeder reactor having a doubling time of not more than 25 years, and established that the major effort and guidance in this program should rest with the Oak Ridge National Laboratory (memorandum from F. K. Pittman to S. R. Sapirie, dated May 18, 1959).

In accepting the responsibility for the development of thorium breeder reactors in its broadest sense, the Laboratory agreed to undertake as one part of the program a thorough evaluation of all types of reactors capable of utilizing thorium by breeding. Although numerous previous studies of thorium breeder reactors had been carried out (5, 26, 22, 43, 44), these studies were either of limited scope, based on out-of-date information or utilized assumptions of such a nature as to make a comparison of the results for different reactors not too meaningful. Thus a Thorium Breeder Reactor Evaluation Group was established in July 1959 to carry out the desired study. This report summarizes the results of the effort to date which has been primarily devoted to an investigation of the fuel cycle potentials of reactor types known to be promising as thermal thorium-breeder reactors.

Five such reactors have been thus evaluated in respect to fuel yield (reciprocal doubling time) and fuel cycle costs. These are: (a) an aqueous homogeneous breeder (AHBR), (b) a molten salt breeder (MSBR), (c) a liquid bismuth breeder (LBBR), (d) a graphite-moderated gas-cooled breeder (GGBR), and (e) a deuterium-moderated gas-cooled breeder (DGBR). An organic-cooled heavy-water-moderated reactor was also investigated briefly; however, this system was dropped because of lack of promise of achieving the specified

maximum doubling time of 25 years. Finally, the fuel cycle costs of a natural-uranium-fueled heavy water reactor of the CANDU type were investigated for the purpose of comparing the relative economics of such reactors and thorium-breeder reactors.

The selection of proper criteria on which to base an evaluation of reactors most suitable for utilizing the energy latent in thorium has long been the subject of controversy. Many people have adopted the premise that a "reasonably high" conversion ratio is sufficient, others that the highest breeding ratio possible should be sought after. The criterion of high fuel yield* has been judged by some to be a better measure of merit than high breeding gain since fuel cycle costs in reactors with a suitably high fuel yield tend to become insensitive to increases in the price of nuclear raw materials. For example, in a reactor with a fuel yield of 4% per year (25-year doubling time), and with 4% annual charges on fissionable material inventory, the value of new fuel produced just offsets inventory charges regardless of the cost (or value) of the fuel. Thus in the very long range picture when fuel inventory charges and net costs of fuel burnup or production may be expected to be dominating economic factors in nuclear power costs, breeders with a high fuel yield will have an economic advantage compared to non-breeders (breeding ratio < 1.0). Under shorter range conditions, however, fuel cycle costs of fabrication, reprocessing, etc. may be as important as fuel inventory and net fuel burnup costs. The achievement of a specific high fuel yield per se, therefore, is not an only criteria on which to base an evaluation of thorium-breeder reactors. Because of this, the criteria of high fuel yield and low fuel costs were adopted as figures of merit for the study herein reported.

The comparative evaluation of breeder reactors, therefore, is based on four major factors: (a) the conversion ratio (breeding ratio) of the reactor, (b) the specific power or efficiency of utilization for fissionable and fertile material, (c) relative costs and difficulty of the fuel cycle, and (d) special features of difficulty or ease of construction and operation

* Fuel yield = $(100 \div \text{Doubling time})$ % per year.

of the reactor. The first two factors determine the fuel yield of the reactor and measure its breeding performance, and factors (c) and (d) are related to the probable cost of power from the reactor and to the length and cost of the development program required to bring the reactor to routine status. Up to the present time, the cost of development and the relation between design complexity and capital costs have been looked at only in a qualitative way; however, Part 2 of the study will examine these issues quantitatively. Since all four factors are strongly inter-related, and for any reactor system any one can be improved at the expense of some of the others, a simultaneous investigation of all of these factors is necessary for a definitive evaluation of thorium breeder reactors.

The first step in the evaluation involved an over-all survey of nuclear reactor systems to establish a list of those which conceivably could satisfy the required criteria of short doubling times and low fuel costs. In this regard, the aqueous homogeneous reactor, by virtue of its high conversion ratio and low inventory of fissionable material, appeared to be most promising in terms of doubling time alone; however, other reactor systems emerged as having fewer fuel, materials, operating and engineering problems and also lower fuel costs. Thus the initial list of reactors to be evaluated included the molten salt reactor, the liquid metal fuel reactor, an advanced version of a gas-cooled graphite-moderated reactor, and a heavy-water-moderated, solid-fuel-element reactor. Reactors not included in this initial evaluation were a thorium fast breeder, an internally cooled molten salt reactor, molten salt burner reactors, a gas-cooled pebble bed reactor, a one-region boiling-slurry aqueous reactor, and a fluidized bed reactor. Although these reactors were judged at this time to be somewhat less promising for meeting the specified criteria, they may have some compensating advantages, such as lower capital costs, greater safety and reliability, and ease of development, which might make them useful to the thorium program. These reactors therefore may be considered later if it seems desirable to do so.

Following the selection of reactor systems to be evaluated, the second step in the study involved the development of a conceptual reactor design for each system, tailored insofar as possible to simultaneously minimize fuel cycle costs and maximize the fuel yield. These conceptual designs were evolved from a review of the available literature by the study group, supplemented by the advice and assistance of persons most familiar with each type of reactor. In the case of the aqueous homogeneous reactor, this approach resulted in a concept utilizing thorium oxide pebbles in the blanket, rather than a slurry as in previous studies. This has the advantages of lower fuel inventory and less fission product poisoning, and the uncertainty in development appears to be less. For the same reasons, the conceptual designs of the gas-cooled, graphite-moderated and gas-cooled, D_2O -moderated reactors also utilized thorium oxide pellet blankets. The molten salt concept selected for evaluation, however, took advantage of the solubility of ThF_4 in molten salt and used a blanket of molten LiF , BeF_2 and ThF_4 . The core of the MSBR was a graphite-moderated bayonet tube design. Finally, the liquid bismuth concept based on a design developed by the Brookhaven National Laboratory utilized a graphite-moderated core and a blanket of thorium oxide slurry in bismuth.

Because of the limited time and manpower available for the study, as well as the lack of knowledge and experience needed to define some of the problems in an analytic manner, a number of arbitrary decisions had to be made based on judgement or "educated guesses". An example of this is the selection of coolant operating temperatures. These were assumed to be as high as practical for each reactor type, and no consideration was given to the influence of temperature on fuel yield. Other major assumptions of an arbitrary nature are: (a) that each station have a net electrical capacity of 1000 Mw provided by at least two reactors, (b) that all processing be carried out in an on-site processing plant, (c) that the reactors are continuously fueled and processed and have reached equilibrium with respect to fission product poisoning and uranium isotopic concentration, and (d) the isotopic composition of new fuel produced is the same as the average composition of the entire system. The bases for these and other assumptions are described in detail in the following section.

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3. ANALYSIS

3.1 General Characteristics of Power Breeder Reactors

A breeder reactor may be defined as a reactor in which a fertile material is exposed to neutrons under conditions such that fissile isotopes are produced in excess of that required to fuel the reactor. The nuclear capability is measured by the fuel yield, or excess fuel produced per year expressed as a percentage of the total station inventory (including that in processing plants) of fissile isotopes and their precursors (e.g., Pa-233). The fuel yield is related to other station parameters by the equation (derived in Appendix I)

$$Y = 37.8 G R P_e F/E I \quad (3.1)$$

where Y is the fuel yield, %/year; G is the breeding gain (excess atoms of fuel produced per atom consumed in the nuclear chain reaction); R is the ratio of fuel atoms destroyed per atom of fuel fissioned* (and having values for the concepts studied ranging from 1.11 to 1.14); P_e is the station net electrical power in Mw (taken as 1000 Mwe); F is the plant utilization factor (taken as 0.8); E is the thermodynamic efficiency (varies from 0.27 to 0.42 in the concepts studied); and I is the total station inventory of fissile isotopes in kg (ranging from 800 to 2000 in this study).

The fuel cycle cost, expressed in mills per net kwhr of electricity produced, is defined as the sum of the costs listed in Table 3.1 less a credit for excess fuel produced.

A generalized flow-sheet for a power breeder reactor is illustrated in Fig. 3.1, where separate fuel and fertile streams are shown passing through a reactor and associated processing plants. A separate coolant stream is also shown; however, in reactors having fluid fuel and/or fertile streams, this would ordinarily be passed through heat exchangers outside the nuclear active portion of the reactor, as indicated by the dotted lines, and an internal coolant would not be used. The reactor may be

* $R = v/\eta\epsilon$

Table 3.1. Components of Fuel Cycle Cost and Rate Charges

Use charges	Rate, %/yr
Fissile isotopes	4
Thorium	12.7 ^a
Fuel and fertile carriers (if any)	12.7 ^a
Replacement charges	
Thorium	Varies ^b
Carriers	Varies ^b
Fuel stream processing plant	29 ^c
Fertile stream processing plant	29 ^c
Breeding credit (at \$15/gram of fissile isotopes)	Varies

a. Interest - 6%, income taxes - 4.6%, local taxes - 2.1%.

b. Not less than 1.3%, corresponding to 30-yr amortization.

c. Interest on capital - 14%, operation and maintenance - 15%.

FIG. 3.1
POWER BREEDER REACTOR

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divided into two regions, viz., a core region containing predominately fuel stream, and a blanket region containing predominately fertile stream. Both streams, though separate, may or may not be present in both regions. Alternatively, the fuel and fertile streams may be combined, in which case the regions would consist of a core and a reflector.

The reactor complex consists of several coupled systems. Those considered in this study are the nuclear, mechanical, thermal, and chemical systems. Others, such as the control and biological protection systems, were excluded because they do not bear directly on the evaluation of the two figures of merit adopted for this study - fuel yield and fuel cycle cost. The fuel yield should be maximized, or, conversely, the fuel cost minimized with respect to all of the independent variables in the several systems over which the designer has any control. The result of such a study may be expressed in the form of a plot of optimized fuel yield vs fuel cycle cost, as illustrated in Fig. 3.2 for hypothetical cases. Each point on the envelope represents a system that has the least fuel cycle cost at a specified fuel yield, or the greatest fuel yield at a specified fuel cost. The envelope-curves may pass through a minimum with respect to fuel cost or may be terminated by some limitation on specific power, processing rate, etc. In any event, the lowest point on the curve is of particular interest, as it is an indication of the economic potential of the concept (relative to design bases, selected unit costs, etc.). In general, the curves will also show a maximum extension to the right. This extreme point is of particular interest, as it is indicative of the economic potential of the system for the long run when fissionable material costs may be greater than those prevailing today.

The curves lying adjacent to the envelope for concept "A" illustrate how that curve is generated. For example, suppose that the yield and cost are strong functions of three independent variables, a, b, and c. These might be the fuel stream processing cycle time, the fertile stream processing cycle time, and the thorium inventory. A small loop is generated by choosing values of a and b and allowing c to vary. The dashed line corresponds to the envelope of such curves, obtained by allowing b to vary while holding a

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FIGURE 3.2

ILLUSTRATIVE REACTOR PERFORMANCE

constant. Changing a would generate other dashed lines, and the envelope of all these is the solid curve for concept A.

In every reactor system there are a great many independent variables at the disposal of the designer. It was not possible to study them all systematically. Instead, it was necessary to eliminate some by arbitrary fiat, some by qualitative reasoning, some on advice of experts, and others by preliminary, exploratory calculations. There remained those variables designated the "key" variables, and these were studied systematically, as far as the limitations on time and effort permitted.

The first problem in the analysis of the four coupled systems is to find sets of specifications that are compatible and practical. Because of the interconnectedness of the systems, this can not be done in a straightforward or "linear" manner; rather, the procedure resembles that of "relaxation" of a network of simultaneous algebraic equations.

3.2 Nuclear System

This system embodies the interaction of neutrons with atoms of fissile and fertile isotopes, moderator, coolant, structure, fission products, and other contaminants. These materials must be grouped in quantities and geometric relationships such that the fission chain-reaction may be propagated at a constant rate, and that the reactor may be cooled and fueled without interfering with the chain reaction. Also, the identity of the various streams (or batches) must be preserved.

3.2.1 Nuclear Economy

The nuclear system is characterized by the breeding gain and the inventory of fissile isotopes. The breeding gain is most conveniently displayed by listing first the mean effective neutron yield, $\bar{\eta}_e$ (defined as the number of neutrons produced from all sources per neutron absorbed in atoms of U-233 and U-235; see also discussion below), together with a listing of various parasitic absorptions and other losses (Table 3.2).

Table 3.2. Nuclear Characteristics

Item	Illustrative Values
Key Variables	
Moderator-fuel ratio, $N(M)/N(U)$	5000 - 8000
Thorium inventory, tonnes	100 - 200
Fuel stream processing cycle time, days	20 - 0.01
Fertile stream processing cycle time, days	100 - 600
Diameter of core, ft	4 - 8
Nuclear Economy	
Net neutron yield ^(a) , $\bar{\eta}\epsilon$	2.210 - 2.250
Nuclear losses ^(b)	
U-233 absorptions	0.900 - 0.920
U-235 absorptions	0.100 - 0.080
Moderator absorptions (including structure)	0.040 - 0.020
Pa-232 absorptions (x 2)	0.020 - 0.040
Th-232 fissions	0.005 - 0.001
Fuel-carrier (or coolant) absorptions	0.050 - 0.005
Thorium-carrier (or coolant) absorptions	0.000 - 0.020
Fission product absorptions	
saturable fission products	0.005 - 0.040
non-saturable fission products	0.040 - 0.010
U-236 and Np-237 absorptions	0.015 - 0.010
Absorptions in miscellaneous materials	0.005 - 0.005
Loss of delayed neutrons	0.003 - 0.000
Leakage from blanket	0.010 - 0.002
Processing losses ^(c)	0.010 - 0.001
Net breeding gain, G	0.007 - 0.100
Other Characteristic Parameters	
Per cent fissions in fertile stream	1 - 35
Per cent thermal fissions	70 - 95
Specific slowing down power, $\xi\Sigma_s/N(U)$	4000 - 30,000

- a. Neutrons produced from all sources per neutron absorbed by U-233 and U-235.
b. Neutrons per neutron absorbed by U-233 and U-235.
c. Fuel atoms lost per neutron absorbed by U-233 and U-235.

The sum of these losses subtracted from the neutron yield gives the net number of new fuel atoms produced per atom of fuel burned, and this number is designated the "breeding ratio" (B.R.). If the value of the breeding ratio is greater than unity, then more fuel is produced than is consumed in the chain reaction, and excess fuel becomes available for sale. The amount of the excess is proportional to $B.R. - 1.0 = G$, the "breeding gain".

In the selection of sets of reactor specifications that optimize the breeding performance, a number of parameters must be considered: moderator-fuel ratio, thorium-fuel ratio, carrier-fuel ratio, and coolant-fuel ratio. The first of these parameters has been investigated by Ergen (83), Chernick (84), Kasten (5), MacPherson (85), and Perry (43, 44, 45). The importance of this parameter stems from the fact that $\bar{\eta}$, the ratio of neutrons produced by fission of fuel atoms per neutron absorbed in fuel atoms, decreases with increasing energy of the neutron absorbed. This ratio has, for U-233, a value of 2.28 (at least) with respect to neutrons ~~having energies up to about 1 ev.~~ (See Appendix F and reference 24). With more energetic neutrons, η falls below 1.9. The behavior of U-235 is similar, except that the value for thermal neutrons is only 2.08, and values as low as 0.81 occur in the resonance regions. Obviously, from the standpoint of production of neutrons, it is desirable for the neutrons to be slowed to energies below 1 ev before they participate in the fission reaction. The specific slowing down power, $\xi \Sigma_s / N(U)$, is proportional to the moderator-fuel atom ratio. Thus, in general, η increases with increasing moderator-fuel ratio.

3.2.2 Neutron Losses

Neutron losses to moderator, contaminants (such as H_2O in heavy water, boron in graphite, etc.) and associated structure (such as zirconium used to contain heavy water) can become appreciable when the ratio of moderator to fuel atoms becomes high. Thus, in general, there is for every system a moderator-fuel ratio such that the difference between the neutron yield and parasitic absorptions associated with the moderator is a maximum. This optimum ratio is not unique for a given moderator-fuel combination, because it depends somewhat on the dimensions of the core (through effect of preferential

leakage of fast neutrons on the spectrum of neutrons), as well as on the concentrations of other absorbers, especially thorium. In the several concepts studied, after the reactor size and composition had been approximately fixed by thermal and mechanical considerations, the moderator-fuel ratio was optimized with respect to neutron yield and moderator captures.

The capture of a neutron by protactinium is especially detrimental in breeders. Not only is the neutron lost, but the product, Pa-234, decays into U-234 instead of U-233, and so an atom of fuel is also lost. The U-234 atoms eventually are exposed to neutrons in the core of the reactor and most of them are converted to U-235 (a small portion, less than 10%, being removed from the system through sale of excess fuel). U-235 has inferior nuclear properties, producing fewer neutrons per neutron absorbed than U-233. Moreover, the side product, U-236, is not a fertile isotope; reaction with a neutron gives U-237, which promptly decays to non-fissile Np-237. Thus, the capture of a neutron by Pa-233 reduces the number of atoms of fuel produced by somewhat more than two.

As shown by Perry, et al (43), when thorium is mixed with the fuel in the core of a one-region reactor (no blanket), losses to Pa-233 are prohibitively high. The losses could be limited by passing thorium through the core separately from the fuel at a rapid rate and then holding it outside the core for a period long compared to the half-life of Pa-233 (about 27 days) before returning it to the core for further exposure. Usually it would be necessary to recycle the fertile material through the core many times in order to build U-233 to levels economical to process. Thus, a large inventory of U-233 would be invested in the fertile stream being held outside the core to allow Pa to decay. This would reduce the fuel yield and increase the fuel cycle cost. It does not appear to be advantageous to have a major portion of the thorium in the core, whether mixed with the fuel or separate.

This conclusion implies that the thorium must be largely segregated into a blanket region. If this is done, then in order to obtain breeding ratios in excess of unity, the dimensions of the reactor must be such that about half of all neutrons can leak from the core region into the blanket

where they can be absorbed by thorium. Thus, as Perry points out (43), at least one dimension of the reactor must be small. Cores in the form of pancakes or of pencils have one short dimension, and provide arbitrarily large volumes.

Losses to Pa are not rendered negligible merely by segregating the thorium in a blanket. The amount of thorium and its disposition are both important. In a blanket loaded with sufficient thorium to make it "black" with respect to neutrons, the rate of production of Pa-233 is independent of further additions of thorium, and depends only on the power of the reactor and the probability of leakage of neutrons into the blanket. Since the rate of decay of Pa is an invariant, and the rate of removal (by processing) is usually small compared to the rate of decay, the amount of Pa in the blanket is nearly constant and independent of the amount of thorium. The Pa, however, must compete with the thorium for neutrons, and the rate of capture of neutrons in Pa is thus inversely proportional to the amount of thorium present, ~~provided the blanket is well mixed and the concentration of Pa is maintained uniform throughout.~~ If the blanket is not mixed, Pa accumulates in regions adjacent to the core where the neutron population is large, and the losses increase. The losses are controlled by mixing, or shifting the fertile material about in the blanket frequently, and by maintaining an inventory of thorium in excess of that required for blackness of the blanket. The operation of shifting increases only the reactor operating cost, but the excess inventory of thorium increases the fuel cycle cost. Also, the fuel yield decreases because of an increase in inventory of U-233 associated with thorium in the fertile stream, and use-costs increase on both thorium and U-233.

Fissions in thorium (by energetic neutrons) make a small but appreciable contribution to the reactivity, and to the effective neutron yield. Defining ϵ as the ratio of the number of neutrons produced from all sources to those produced from U-233 and U-235, the effective neutron yield (neutrons produced per neutron absorbed in fissile atoms) is given by the product $\bar{\eta}\epsilon$, where $\bar{\eta}$ is the mean value for the mixture of fuel isotopes (averaged over the neutron spectrum). The number of fissions in thorium,

however, must be subtracted from the total absorptions in thorium when the yield of new fuel atoms produced is computed.

In the solid fuel reactors studied, neutron losses to fuel-carrier (graphite) were small and special effort to reduce them was not worthwhile. The ratio of carrier to fuel was conditioned rather by the need for sufficient heat transfer surface, and the need to control thermal stress and radiation damage in the fuel matrix.

The thorium carrier was appreciably parasitic on the neutron economy only in the MSBR and LBBR concepts. Maximum concentrations of thorium were used. In the other concepts, the fertile stream consisted of pellets of ThO_2 , and the oxygen absorbed a negligible number of neutrons. In the AHBR, the blanket structural material (Zircaloy) absorbed appreciably.

In both the solid fuel reactors reported herein, the coolant was helium at ~ 2000 psi. Helium absorbs neutrons negligibly; however, the effect of the nearly empty coolant channels on the nuclear performance must be considered. These voids reduce the mean density of the core, and tend to increase the leakage of neutrons to the blanket. More important, axial "streaming" may increase end leakage of neutrons and place a greater burden on the end-blankets, which are apt to be inefficient neutron absorbers because of the requirements for transfer of coolant and fuel elements.

Neutron losses to fission products are important in all concepts. Fission products may be divided into two groups - those removed by processing the fuel isotopes and those which may be removed by other means. The second group comprises xenon, krypton, and other less important gases, and, in the AHBR, certain insoluble rare earth oxides.

In the AHBR, it has been demonstrated that gaseous fission products may be removed by stripping radiolytic gas from the solution. In the MSBR and LBBR, however, the fuel is in direct contact with graphite which will probably not have a density greater than 1.8 g/cc . Since the maximum density is 2.25 g/cc , the proposed grade contains pores and interstitial voids in excess of 15% of the volume. Unless the surface is rendered impervious by some means, xenon may diffuse into and be retained in these spaces until reaction with neutrons takes place.

In the MSBR concept, the fuel is contained in bayonet tubes constructed of graphite (Sec. 5.2.3), and the surface of graphite exposed to the fuel is relatively small. By modifying the treatment given by Watson and Evans (147), the loss of neutrons to xenon was calculated as reported in Appendix B. It appears that, if the coefficient of diffusion of xenon in graphite is no greater than 10^{-7} cm²/sec, the xenon poison fraction in the MSBR can be held to 0.005 by sparging the fuel with helium on a 55 sec cycle. Since the transit time of the fuel through the reactor, pump, heat exchangers is about 3 sec, only about 6% of the circulating fuel stream, or about 10 ft³ per sec, need be passed through the sparging chamber. Furthermore, it may be that some considerable portion of the xenon can be removed by sparging the fuel in the pump bowl above the impeller.

According to Watson (147), the selected value of the diffusion coefficient is not unreasonable for reactor grade graphite. However, even if the coefficient is as large as 10^{-5} cm²/sec, as presently reported for some grades of National Carbon CEY graphite (147), the xenon poison fraction can be held to 0.005 by sparging the fuel at the rate of no more than 20 ft³/sec (Appendix B), provided the removal efficiency is near 100%.

In the LBBR concept proposed by Thomas, et al (22), and studied here, the surface of graphite exposed to the fuel stream is much larger than in the MSBR, and since the solubility of xenon in bismuth is about the same as in the molten salt, the poison fraction will probably be prohibitively large. However, some version of the bayonet tube construction could probably be developed for the bismuth reactor concepts, and the xenon poison fraction controlled satisfactorily by sparging.

Thus, it appears reasonable that xenon can be removed from the fuel stream of the MSBR and the LBBR by sparging them with helium in the surge volumes above the pump impellers (Appendix B). The possibility that in the solid fuel reactors xenon might diffuse out of unclad fuel elements with sufficient rapidity to effect a saving of neutrons is presently under study.

In circulating fuel reactors, the loss of neutrons to fuel-carrier can be reduced by increasing the concentration of fuel atoms in the carrier. In the AHBR, this can be compensated by decreasing the diameter of the core; in the MSBR and LBBR, the volume fraction of fuel-stream in the core can be reduced. In either case, the mass of carrier exposed to neutrons is reduced. In the AHBR, since the carrier (D_2O) is also the moderator, this change involves a change in the moderator-fuel ratio, whereas in the other two concepts, where the moderator is graphite, this ratio is not changed. The three circulating fuel systems have this in common, however, that increasing the concentration of fuel in the carrier results in an increase in the inventory of fissile isotopes in that part of the circulating system lying outside the nuclear-active zone (i.e., in pumps, heat exchangers, external piping, etc.). This increase impairs the fuel yield and increases the fuel cycle cost.

Rare earth oxides may be conveniently and economically removed from the fuel stream of the AHBR by means of small cyclone separators ("hydroclones"), and the loss of neutrons to these isotopes (comprising a large fraction of total fission products), can be held to very low levels (Appendix E).

Other fission products remain associated with the fuel, and are removed only by removing the fuel and treating it in a processing plant. Rapid processing tends to improve the fuel yield and increase the breeding credit, but the improvement is limited by increased inventories, processing losses, and costs, and the net fuel cycle cost eventually increases.

The reactor designer has little control over loss of neutrons to U-236, other than through control of Pa absorptions, as discussed above. Likewise, loss of delayed neutrons in the circulating fuel reactors is fixed by heat removal requirements. Leakage is controlled by using enough thorium to make the blanket "black", or by providing a reflector outside the blanket. Processing losses are peculiar to the scheme employed, and proportional to the processing rate.

3.2.3 Calculations

The nuclear system was calculated by means of the GNU 34-group, one-dimensional, multi-region program for the IBM-704 (85). The input consists of specifications of reactor geometry, dimensions, and the mean densities, $(\bar{N})$, of the atomic species in the various regions. These latter are necessarily "homogenized" over a region. For example, the mean concentration of U-233 associated with the fuel-stream in the core was calculated by dividing the mass of U-233 in the fuel stream inside the core by the total volume of the core region.

The GNU program provides a "criticality search" by means of which either a dimension or one or more concentrations is varied until the multiplication constant, k , differs from unity by less than some specified small amount. The output consists principally of the fraction of neutrons, A , involved in capture or fission reactions with each atomic species in each region. Thus,

$$A_{i,j,k}^s = \bar{N}_{i,j,k} \sum_u \sum_{V_k} \bar{\phi}(\Delta u) \sigma_i^s(\Delta u) \Delta u \Delta V_k \quad (3.2)$$

where s denotes type of interaction (n , γ or fission), i denotes atomic species (U-233, U-234, Pa, etc.), j denotes the stream (fuel or fertile), k denotes the region (core or blanket), and other symbols have their customary meaning. It should be noted here that in three of the reactors studied (MSBR, LBBR, and DGBR), part of the fertile stream passed through the core, and in two (MSBR and LBBR), the fuel stream passed through end blankets. Thus, "fuel stream" and "fertile stream" are not synonymous with "core" and "blanket".

The quotients $A_{i,j,k}^s / \bar{N}_{i,j,k} = C_{i,j,k}^s$ are reaction coefficients having useful property that, when multiplied by the mean atomic concentration, $\bar{N}_{i,j,k}$, of the corresponding element (i -th element of j -th stream), the product is equal to the fraction of neutrons involved in interactions of the s -kind with that element in the k -th region. Further, if this product is multiplied by F , the number of neutrons produced per unit time in the reactor, the new product, $\bar{N}_{i,j,k} C_{i,j,k}^s F$, is the rate of neutron reactions of the s -kind with element i , j in region k .

Equilibrium calculations (Appendix H) were performed by means of the ERC-5 program (86) for the IBM-704, which employs the coefficients C along with other specifications, such as reactor power, fuel volume, fertile stream volume, processing rates, etc., to compute the equilibrium concentrations of Pa, uranium isotopes through U-236, Np-237, xenon, Sm-151, Sm-149 and "other fission products". The constraints imposed on the system of equilibrium equations are the conservation of mass, maintenance of criticality, and distribution of neutrons. The latter constraint was satisfied by accounting for all neutrons, including those absorbed by thorium, moderator, carriers, etc., and those leaking out of the blanket by means of "flux factors" (Appendix H).

The principal assumption involved in this treatment is that the coefficients C are insensitive to small changes in concentration of important elements (U-233, U-235, fission products) and to large changes in the concentrations of minor elements (U-234, Pa-233, xenon, etc.). In those cases where the equilibrium calculation produced new concentrations sufficiently different from those used in the initial GNU case that the corresponding change in the coefficients would be appreciable, it was necessary to repeat the GNU calculation to obtain new coefficients, etc.

Item No.	Elements Used in ERC-5	
	Fuel Stream	Fertile Stream
1	Thorium	Thorium
2	Protactinium	Protactinium
3	Uranium-233	Uranium-233
4	Uranium-234	Uranium-234
5	Uranium-235	Uranium-235
6	Uranium-236	Unassigned
7	Neptunium-237	Unassigned
8	Other F.P.*	Other F.P.*
9	Xenon-135	Xenon-135
10	Samarium-151	Samarium-151
11	Samarium-149	Samarium-149
12	Carrier	Carrier
13	Moderator	Leakage and delayed neutrons

* Fission products other than Xe, Sm.

The above procedures sufficed for the calculation of "homogeneous" reactors, e.g., those in which the fuel and moderator are intimately mixed (AHBR, GGBR). Additional steps were required for the "heterogeneous" reactors (LBBR, MSBR, DGBR). After a preliminary equilibrium calculation, a lattice-cell calculation was performed by means of the Cornpone program for the Oracle (91). For example, in the MSBR calculations, the unit cell consisted of an inner cylinder of fuel stream surrounded successively by annuli of graphite, fuel stream, graphite, fertile stream, and moderator graphite. The boundary conditions were zero net radial current at the periphery and zero axial buckling. The program computed the ratio of the mean flux intensity in each sub-region of the cell to the average intensity in the cell -- these ratios were designated "disadvantage factors".

$$D_{i,j,n} = \frac{\frac{1}{V_r} \int_{V_r} \phi_n \Delta v}{\frac{1}{V_c} \int_{V_c} \phi_n \Delta v} \quad (3.3)$$

where $D_{i,j,n}$ is the disadvantage factor associated with i -th element in the j -th stream for lethargy interval n , V_c is the volume of the cell, V_r is the volume of the sub-region, and ϕ_n is the n -th group mean flux in dV .

These disadvantage factors are employed in 34-group-diffusion calculations of the corresponding finite reactor. The "homogenized" concentration of each isotope is multiplied by the appropriate disadvantage factor so that all properties that are dependent on atomic density of that isotope are weighted by the relative flux to which the nuclei are exposed. For example, absorptions in a given species are computed thus:

$$A_{i,j,k}^a = \sum_{V_i} \sum_n D_{i,j,k,n} \bar{N}_{i,j,k} \sigma_{i,n}^a \bar{\phi}_{i,n} \Delta v_n \Delta V \quad (3.4)$$

where $\bar{\phi}_{k,n}$ is the n -th group mean flux in a "homogenized" increment of volume ΔV .

The coefficients C , defined on page 27, are thus "disadvantaged" and reflect the heterogeneity of the fuel and moderator mixture. The equilibrium calculation on ERC-5 then proceeds as before.

The nuclear data used were compiled by C. W. Nestor (24, and Appendix F). The information on η of U-233 available in the Fall of 1959 was used to estimate group cross sections. The corresponding group-average values of η are given in Appendix F.

Allowance for resonance saturation (self-shielding) was made only in the case of thorium, and here Doppler broadening of the resonances was also taken into account. The results of this study are not sensitive to even moderate errors in effective thorium cross section for the reason that the performance calculated for some given concentration can always be achieved in practice at some neighboring concentration. Thus, errors in the cross section of thorium appear, in the result, as modest errors in the thorium inventory, and these are of minor importance.

Composite cross sections for "other fission products" (excluding xenon and samarium) were estimated from the work of Pattenden (79). The 2200 m/s cross section was 50 barns per fission and the resonance integral was 138 barns. The 2200 m/s cross section of Pa-233 was taken as 70 barns, and the resonance integral as 1200 barns. The 2200 m/s cross section of reactor grade graphite was taken to be 4.2 mb, which allows 0.8 mb for contaminants such as boron. Other aspects of the nuclear calculations are taken up in Appendices D and E or in the sections devoted to specific reactor concepts.

3.3 Mechanical System

The mechanical system is concerned with the arrangement, support, and motion of the various reactor components. These include the moderator, fuel stream and coolant (if different from fuel carrier), core vessel, fertile stream and coolant, reflector, thermal shield, and reactor vessel. The analysis of the mechanical system included the estimation of the limits on size and operating pressure in the various reactor vessels (Appendix A). The economic aspects of these and related parameters were studied by Perry (87), who prepared a schedule of limiting pressure vessel thickness vs diameter. He concluded that thicknesses up to 10 in. could be achieved with vessels up to 14 ft in diameter. Consideration was also given to the design of the fuel elements for the GGBR and the DGBR. No work was done on the conceptual design of the fuel element handling machinery for these reactors, and only brief consideration of the structure of the cores.

In general, it was found that the mechanical systems of the several reactors were loosely coupled to the other systems, and could be approximately designed without regard to them. The principal exception to this was in the coupling between pressure vessel diameter and heat transfer in the core in the two gas-cooled reactor concepts (GGBR and DGBR). As shown in the next section, the heat removal rate in the core of a gas cooled-reactor is directly proportional to the pressure. The pressure equation (Appendix A) is

$$P = \frac{2SE_w T_v}{D_v + 1.2 T_v} \quad (3.5)$$

where P is the operating pressure (psig), S is the allowable stress in the pressure vessel (psi), E_w is the weld efficiency ($E_w \leq 1.0$), T_v is the thickness of the pressure vessel (ft), and D_v is the inside diameter of the pressure vessel (ft). This last is fixed by summing the thicknesses of the core, blanket, reflector, and thermal shield. In the several concepts studied, vessels in the range from 10 to 13 ft in diameter appeared to suffice and these may be shop-fabricated and shipped by rail.

Details of a method of estimating the required thickness of thermal shields was worked out, but it was concluded that a nominal allowance of 12 in. would be adequate for all designs.

3.4 Thermal System

The thermal systems in the solid fuel and fluid fuel reactors require different treatment. In the solid fuel reactors, there is the problem of providing for adequate heat transfer between fuel and coolant inside the reactor without impairing the nuclear performance or complicating the mechanical system. Outside the reactor, however, the heat transfer system is almost completely decoupled from the other three systems, and is easily designed and optimized. The only coupling is through the work expended on pumping the coolant through the external circuit and the relation of the thermodynamic efficiency of the steam-electric plant to the temperature drops in the primary heat exchangers.

In fluid fuel reactors, the fission heat is largely released in the fuel stream and there are only minor heat transfer problems (e.g., cooling of moderator and structure). There may be limitations on the power density, as in the AHBR, where it is limited by requirements for maintaining the integrity of the core vessel wall, or as in the LBHR, where it is limited by a maximum allowable velocity. But these are not problems of heat transfer between distinct phases. On the other hand, heat must be transferred in external heat exchangers with rather narrow temperature differences between maximum and minimum fuel-stream temperature, and with rather small differences between mean fuel-stream and primary coolant temperature, else the thermodynamic efficiency suffers. These requirements must be met using heat exchangers, pumps, piping, etc., having volumes as small as possible, and without exceeding limits on velocity or pressure drop.

Detailed analysis of the heat transfer in the external circuits of fluid fuel reactors were performed by I. Spiewak and L. F. Parsly (11). An extract from their work is given in Appendix A. They established a number of criteria important in the design of such systems, and were able

to optimize the design, to a considerable extent, with respect to external power density. This quantity was used directly, in this study, to couple the nuclear, chemical, and thermal systems, for it fixes the volume of the external fuel circuit as a function of heat rate in the fuel system (which varies from case to case). The volume is a parameter in the equilibrium equations, and relates to the costs on the one hand and to the distribution of neutron captures on the other.

Estimated Power Densities in External Circuits
of Three Fluid Fuel Reactors (11)

Reactor Concept	Fuel Carrier	Power Density Mwt/ft ³
AHBR	Heavy water	1.5
MSBR	Molten salt	7.6
LBBR	Liquid bismuth	3.4

Molten salt appears to be a superior heat transport fluid; however, the degree of merit of a fluid fuel-carrier is also related to the fuel inventory. This varies from case to case, as discussed in Sec. 3.2, but typical values give specific powers in the external circuits of 8.0, 3.3, and 2.9 Mwe/kg U for the AHBR, MSBR, and LBBR respectively. The specific power was based on electrical output because the overall plant efficiency is also a function of the fuel carrier. On this basis, D₂O is clearly a superior fuel carrier, in spite of the low thermodynamic efficiency associated with its use.

Limitations on power density inside the cores of the fluid fuel reactors are discussed in the sections devoted to the individual reactors. These were not related in any way to consideration of temperature rise due to generation of after-heat during a loss-of-pump accident, since control and safety problems were excluded from the study. Attention was given, instead, to the problem of minimizing the volume of the fuel stream inside the reactors. The neutron-active volume is fixed, of course, by nuclear considerations. But the volumes of fuel invested in downcomers, plenum chambers, etc., are fixed by consideration of core size, fuel velocity, pressure drop, etc. These are discussed under the individual concepts.

The thermal system of the gas-cooled, graphite moderated reactor selected for study (GGBR) is analyzed in Appendix A. Similar studies on related reactor concepts have been performed by Ozisik and Korsmeyer (88, 89), Cottrell, et al (62, p. 36), and by Carlsmith (50), whose study is directly applicable to the case in hand.

The power equation has several forms, of which the following is an example.

$$Q = \frac{P d^2 [C_p (T_2 - T_1)]^{3/2}}{T_1 (1 + T_2/T_1)^{1/2}} \left[\frac{R_1 R_2 F_1^3 t_2}{R_3^f F_2 L} \right]^{1/2} \times (C.F.) \quad (3.5)$$

where Q is the heat rate, P is the pressure, D is the diameter, T_1 is inlet (absolute) gas temperature, T_2 the outlet temperature, R_1 the ratio of pump-work to reactor heat, R_2 the fraction of pump-work expended in core, F_1 the fraction of core volume occupied by gas passages, F_2 the fraction occupied by fuel (plates), t_2 the thickness of the fuel plates, f the Fanning friction factor, L the length of the gas passages in the core, and C.F. represents conversion factors and constants such that the variables may be given in familiar units.

Values of the variables in Eq. 3.4 may not be assigned entirely independently, as there are various factors which limit their ranges. For instance, the length L may not be made indefinitely short because of nuclear considerations. Also excessive circuit pressure drops, ΔP , must be avoided. As shown in Appendix A,

$$\Delta P = \frac{P R_1 C_p (T_2 - T_1)}{R T_1} \times (C.F.). \quad (3.6)$$

Of the independent variables appearing in this equation, it is seen by Eq. 3.5 that the heat rate, Q, is least sensitive to R_1 , the ratio of pump-work to reactor heat rate. Accordingly, an upper limit is placed on R_1 by considering the allowable pressure drop, and this is limited

by a consideration of dynamic forces imposed on the core structure and on the pump impeller and bearings, etc.

The gas-film temperature rise (Eq. A-13, Appendix A) may conveniently be controlled by adjusting F_2 , the fraction of core volume occupied by fuel plates, as Q is insensitive to changes therein. Fuel-plate thickness must lie in a range fixed by mechanical strength requirements and thermal stress limitations (Eq. A-21, Appendix A). Also, the maximum allowable fuel-plate temperature may limit fuel plate thickness.

In general, there is no straightforward procedure for generating optimum and compatible sets of specifications. The designer must become familiar with the characteristics of individual reactor concepts and develop skill in rapidly selecting design specifications that will optimize the performance and satisfy all the requirements simultaneously.

3.5 Chemical System

The chemical system comprises the fuel and fertile stream processing. Analysis consists in estimating the rate of removal of various isotopes from these streams as a function of cycle time, in estimating loss of fissile isotopes to waste, and estimating the hold-up of fissile isotopes in the various stages as a function of processing rate. The removal rates and losses are used in the equilibrium calculations referred to in Sec. 3.2, the inventories enter into the calculation of the fuel yield and the fuel cycle costs.

The various processes used in this study are discussed in detail in Appendix C. Sometimes the analysis is straightforward, as that of the Thorex process, where the removal efficiency for Pa, U-233, and fission products is 100%. In other cases, the streams may be treated by more than one process, each selectively removing a certain group of isotopes. For instance, in the MSBR, U-233 is removed from the fertile stream by the fluoride volatility process, but Pa-233 presumably is not. Fission products are removed only by discarding the fertile stream salt carrier. In the MSBR fuel stream processing, uranium isotopes are removed by fluoride volatility, rare earths by HF dissolution, and other contaminants by discarding the salt carrier.

It should be noted that the composition of the mixture of uranium isotopes in the fuel stream is not directly related to the fuel stream processing rate, as uranium is removed from the system only by sale of excess fuel. This, of course, is ultimately related to the processing rate through the poison fraction, but the coupling is through the equilibrium equations and the critical equation. There is a certain amount of amplification in the coupling in that the lower the poison fraction the greater the sales and corresponding greater rate of removal of U-234, U-235, and U-236. This shifts the composition of the mixture in the reactor toward U-233, which improves the neutron yield, $\bar{\eta}\epsilon$, which increases the breeding gain further and also reduces the rate of formation of the higher isotopes of uranium, etc.

The inventory of U-233 associated with the decay of Pa-233 in a Thorex plant depends upon the number of times the mixture of fission products and partially decayed protactinium is reprocessed to recover the U-233 formed, and decreases as the number of reprocessings increases (81). However, it was concluded that it would not be economical to reprocess the fission products more than once (Appendix C). For one recycle, it is easily shown that the break-even holding time is about 200 days when the value of fissile isotopes is \$15/gram and the use charge is 4% per annum.

3.6 Accuracy and Precision

In the calculation of the nuclear economy, the effective neutron yield was computed to five significant figures. This precision exceeds the accuracy with which most of the data are known to a considerable extent, but was necessary because breeding gains range down to 0.0200, and four decimal places are required to give a "relative" accuracy of three significant figures in the breeding gains for a given reactor concept. Comparisons between different reactor concepts cannot be made on this level of accuracy.

4. CONCEPTUAL DESIGN AND
COST BASES

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4. CONCEPTUAL DESIGN AND COST BASES

An attempt was made to select design bases that would not favor any particular reactor concept, that would be consistent with the development of a substantial nuclear power industry, and that would permit estimation of fuel cycle costs using current price schedules.

For those concepts which initially appeared to be less promising than the best thorium breeders based on previous studies, it was elected to allow a greater degree of optimism in regard to possible design improvements. This approach, it is believed, eliminates the possibility that an early bias in favor of certain systems influences the relative merits of all of the systems considered.

4.1 Design Bases

4.1.1 Capability of Electrical Plant: 1000 Mwe

~~Nuclear reactor power plants built or in construction as of June 30, 1960 include one each having capabilities of 255, 180, and 110 Mwe, and eight below 100 Mwe. Those planned include one at 360 Mwe, and seven below 100 Mwe. The AEC has published several studies in which the reference design plant capability was ranged up to 300 Mwe (131, 132). However, it appears, from the many studies that have been performed, that these small nuclear power plants cannot compete economically with conventional plants.~~

Presently, about 11% of the power from stationary plants generated in the United States is produced in 1000 Mwe or larger stations, and the fraction generated in the larger stations is increasing rapidly. By 1970, new installations, having capabilities of 2500 Mwe, or more, will probably be common. Since the costs bases used herein are those presently prevailing and since the power costs are to be compared to those of current conventional plants, 1000 Mwe was selected for the capability of the nuclear power plants to be studied. Also, it appears that chemical plants associated with electrical plants of this size are about at the break-even point with respect to on-site vs central location, and no extra cost penalty results from the use of on-site plants (17).

4.1.2 Fuel Reprocessing: On-Site

The inventory of fuel in transit and the allocation of fuel in process in a central processing plant depends upon the number of stations served, their average distance from the central plant, and other factors difficult to evaluate. The associated costs are equally ambiguous, as are also the losses. On-site plants, on the other hand, are susceptible to reasonably precise analysis of inventories, costs, and losses. Further, in the fluid fuel reactors, the processing rates required to achieve the highest performances are so great that it is clearly necessary and economical to couple the reactor to an on-site processing plant. For example, in the AHBR, the cycle time for treating the entire circulating fuel volume in the hydroclones, by means of which insoluble rare earth oxides are removed from the fuel stream, is about 15 minutes. In the MSBR, uranium isotopes are removed from the fuel stream by fluorination. This process would clearly be done on-site in order to avoid charges on an inventory of fuel in transit even though the stripped salt were shipped elsewhere for removal of fission products.

4.1.3 State: Equilibrium, Steady, Symmetrical

In a breeding economy, where excess fuel is used to start new reactors, the reactors have, initially, low concentrations of fission products in the fuel stream, and no protactinium in the fertile stream. This initial advantage is offset by unpredictable start-up inefficiencies and shake-down losses. Further, the approach to equilibrium occurs in a time short compared to the doubling time of the system, which, in the thermal reactors studied here, is not less than five years, whereas the relaxation times for Pa and fission product buildup are typically of the order of 40 days. Thus, even at the shortest doubling time, fewer than 5% of the reactors in a doubling system will have been on the line less than two relaxation times. For these reasons, the initial non-steady period of performance was neglected, and all reactors were computed in the equilibrium state.

The fluid fuel reactors operate naturally in a steady (as opposed to a cyclic) manner. Additions to and withdrawals from the fuel and fertile streams are done batchwise, but the amounts are so small compared to

the inventories, and the frequencies so great compared to the processing cycle frequency, that the processing may, for purposes of calculation, be treated as continuous and concentrations in the various streams as constant.

Presently operating solid fuel reactors, and many of those being built are operated in a cyclic manner; i.e., the reactor is loaded with fresh fuel and the excess reactivity is neutralized by means of control elements containing nuclear poisons. As the fuel burns out and fission products accumulate, the control elements are gradually withdrawn. When the excess reactivity is exhausted, the reactor power is reduced to zero, coolant flow is stopped, and the spent fuel is discharged. During the cycle, the breeding ratio, and the distribution of fissions, temperature, etc., vary widely. Moreover, the fuel may be recharged several times while the fertile stream is undergoing one cycle of exposure to neutrons. As U-233 accumulates in the fertile stream, it will contribute to the fission rate. Thus, every cycle of the fuel stream will be different within the longer cycle of the fertile stream.

The computation of the average performance of such a periodic system is both difficult and tedious. Different approximations, assumptions, and methods of computation could lead to significantly different results.

Unless exposure limits can be increased proportionately with power density, cycle times will become shorter as the art advances, and batch-loaded reactors will spend a greater proportion of time in the "down" condition. And, regardless of exposure limits, considerations of safety always limit the excess reactivity that can be used, and so limit the cycle time. For these reasons, the continuous fueling of solid fuel reactors at full power is growing in favor (106). It is expected that then the plant utilization factor will be determined by the ratio of peak-to-average load, and that maintenance can be done during periods of minimum load.

There are other, nearly equal benefits that accrue from the use of continuous fueling. The reactor operates in a steady state, and therefore can be designed to operate at peak performance continuously, whereas a cyclically fueled reactor can be designed to operate at peak performance only at one point in its cycle, and the average performance will be lower.

The uncertainties in the design of steady state systems are considerably less, and safety factors need not be as large. Further, only reactivity in excess of the amount needed to control the power level need be provided, and this is much less than that required to enable batch-loaded reactors to achieve high burnups. The saving of neutrons is appreciable.

It was concluded that there are strong incentives, especially in the design of power breeder reactors, to use continuous fueling techniques, and though these are today only in initial stages of development, it was assumed that they will become available. The solid fuel reactors were evaluated on this basis.

Recently, W. B. Lewis (133), among others(106), has proposed a bi-directional, continuous loading technique which has many advantages. Fresh fuel elements are loaded at both ends of a reactor and move through the core countercurrently in adjacent passages. As shown in Appendix G, this mode of operation results in nearly uniform concentrations of fuel isotopes and fission products when these are averaged over a cross section perpendicular to the axis of motion. The rate of travel in a channel is made proportional to the average fission rate in that channel so that the "burnup", or fraction of fuel atoms fissioned, will be the same in every channel. This makes the distribution of neutron population, in cylindrical reactors, symmetrical about the midplane. As a result, the ratio of peak-to-average power is less than if the reactor were fueled at one end only.

4.1.4 Composition of Product: System Mixture

The operator of a breeder reactor may elect to sell almost pure U-233 from the blanket stream, or he may recycle all of the freshly bred material and sell spent fuel exclusively. The equilibrium state of the reactor will, of course, be different for the two modes of operation. If only U-233 is sold, then higher isotopes of uranium (U-234, U-235, and U-236) have no way of escaping from the fuel system, and remain till they are fissioned or transmuted. The chain ends with Np-237, which is presumably removed by any processing method that also removes rare earth fission products. The contribution of U-235 to the fission rate results in a lower η for the system and greater loss of neutrons to capture in U-236. On the other hand, the value of the product U-233 is presumably high, since it would give

superior performance during the start-up transient of a new reactor.

In contrast, the sale of spent fuel only would tend to purge the reactor of the higher isotopes of uranium, and would result in a higher breeding ratio and production rate. The value of the product, however, would be less than that of U-233, since its use to put a new reactor on the line would result in an inferior initial performance.

It is clear, however, that the higher isotopes of uranium are "burned" sooner or later in some reactor of the system, and that the overall performance can vary little, if at all, with the mode of operation. Hence, the product of the unit value of the excess fuel times the rate of production is probably independent of the mode of operation. Similarly, the value of the annual product divided by the value of the inventory, is probably independent of the mode of operation.

If these hypotheses are accepted, then it becomes convenient to ~~evaluate that equilibrium state of the reactor corresponding to a mode~~ of operation wherein the composition of the product, obtained by mixing portions of the fertile stream with portions of the spent fuel, is the same as that of the average of the system; i.e., that obtained by mixing all the inventories of fuel in the system, including dump tanks, processing plants, etc. When this is done, the product has the same unit value as the average unit value of the plant inventory, and the fuel yield is ~~unambiguously defined as the ratio of kg of excess fuel produced per year~~ to the kg of fuel isotopes in the plant.

The evaluation can be done in this way, even though in practice another mode of operation might be used. In order to produce a system mixture, either two processing plants are required (one each for the fuel and fertile streams), or else a single processing plant is alternately fed with one stream while the other is accumulating. Either way, the capital investment would probably be greater than that of a plant which was fed continuously with a stream consisting of a mixture of the fuel and fertile streams. The latter scheme is undoubtedly the one that would be used.

4.1.5 Fuel Inventory: All U-233, U-235, and Pa-233 in Plant

When a new reactor is started up, only fuel sufficient to fill the fuel circuit is required. The amount ranges, in the reactors studied, from 15 to 60% of the total plant inventory. Reactor plants started in this way do not produce excess fuel at the maximum rate until the inventories of fuel in the fertile stream, processing plants, reserve, etc., have reached their equilibrium values. This loss in production during start-up results in the investment of more operating capital than would be required if the entire plant were inventoried at the start and production of excess fuel begun immediately. If the interest charges on operating capital are greater than the use charges on fuel, then starting with a complete inventory would be economically advantageous.

In any event, the complete inventory must be provided, either initially, or during a start-up transient, and the calculation of the fuel yield, in per cent per year, must be based on it, rather than on the partial inventory on the fuel stream. In this calculation, Pa-233 counts as a fuel isotope. Although it is not present in the blanket of a new reactor, if an equivalent amount of U-233 is not provided, an equal amount of production will be lost while the protactinium is accumulating in equilibrium amount.

4.1.6 Number of Reactors per Station: Two or More (Integral Number)

It was considered unwise to specify fewer than two reactors for stations of 1000 Mwe capacity for the reason that if only one is used it would be difficult to achieve a plant utilization factor of 0.8. With two reactors, the entire load can be handled by one of them during some minimum load periods, and during average load periods, one reactor could supply about 60% of the load. Maintenance would be performed during minimum load periods whenever possible.

In all concepts studied, some one degree of freedom was found by means of which the number of reactors required per station could be adjusted to an integral value without affecting the performance of the system as a whole, as measured by the fuel yield and the fuel cycle cost.

In the AHBR, for example, the length of the core and the concentration of fuel in the fuel stream can be adjusted with little disturbance of the rest of the system. In the GGBR, power from a single core can be adjusted by varying the thickness of the fuel plates and the redistribution of available graphite between moderator-structure and fuel plates. In the DGBR, the number of cores can be reduced to an integral number by flattening the radial distribution of fission rate. This can be done by making the lattice pitch a function of radial position in the manner described by Goertzel and Loeb (90). In this method, the lattice spacing is made smaller for those channels lying adjacent to the axis of the core, and larger for those lying near the periphery. The central region of the core is thus undermoderated and fast neutrons tend to leak out to the peripheral regions, which are overmoderated. The thermal flux and thus also the fission rate distribution can be made nearly uniform. Consequently, the power produced in each channel is the same, the cooling-gas flow rate is uniform, and every channel can be operated at peak capacity.

In the MSBR and the LBBR, the adjustment to an integral number of cores can be made by relaxing the limits on the fuel-stream velocity slightly. Nominally 20 fps, it was allowed to vary by 10% as required.

4.1.7 Fuel Reserve: Thirty-Day Supply

From the standpoint of plant utilization, it would be disadvantageous to halt the production of power for lack of fuel resulting from an emergency or scheduled stoppage in the processing plant, and a supply of fuel would be kept at hand sufficient to keep the reactor operating at power for some nominal period independently of the operation of the fuel processing plant. A reserve sufficient for thirty days of operation was selected for this study.

This design basis affects fluid and solid fuel reactors differently. In the fluid fuel reactors, fuel withdrawals may be stopped when the processing ceases. In order to maintain operation of the reactor, it is then sufficient merely to add fuel to replace that consumed in the chain reaction plus a small increment to override the effects of fission product

accumulation. In the solid fuel reactors, however, when fuel elements reach the discharge end of the core, they must be removed from the reactor and replaced by fresh elements. The fueling rate is not changed, and the reserve consists of a 30-day supply of fresh fuel elements. For reactors having burn-ups of 25%, the reserve will thus be approximately four times larger than in a fluid fuel reactor plant having the same thermodynamic efficiency.

4.1.8 Number of Regions: Two

Previous experience indicated that it would be difficult, except in the case of an aqueous reactor, to design one region reactors (core and reflector, with fertile material mixed with fuel) which would satisfy the requirements of the directive on which the study was based -- namely that the systems investigated should show promise of being able to achieve fuel yields of 4% per year or more. For this reason, consideration of one region reactors was excluded from the present study.

4.1.9 Plant Utilization Factor: 0.8

The customary value of 0.8 was assumed. A minimum of two reactors per station, as discussed above, was specified in consistency with this assumption. Also, a reserve supply of fuel sufficient to maintain the power generation plant in operation for 30 days with the processing plant shut down was specified.

4.1.10 Fuel Cladding: For Compatibility Only

Disadvantages of using clads on the fuels are several. Cladding increases the cost of fabrication, frequently increases parasitic captures of neutrons, and impedes the flow of heat from the fuel to the coolant.

It was decided that the use of a special clad on solid fuels or on thorium oxide pellets would not be specified except to achieve chemical compatibility between these and their coolants. None was required for the graphite-dispersion fuels cooled by helium or for the thorium oxide pellets cooled by heavy water or by helium. These coolants are therefore in a

"dirty" condition, and there are objections to this from the standpoint of maintenance. However, the situation is no different in regard to the maintenance of the circulating fuel streams in the fluid fuel reactors, and the use of dirty cooling systems would appear to put the various systems on more comparable bases.

In the LBBR and the MSBR, it was assumed that the carrier fluids (bismuth and molten salt) could be used in direct contact with the moderator (graphite). The evidence in support of these assumptions is discussed in the individual sections devoted to these reactors.

4.1.11 Waste Disposal: No Estimate

Methods of waste disposal and the associated fuel cycle costs were not considered in this study for the reason that the problems and the cost will be roughly the same in every reactor concept, and the selection of a concept for development will not be based on the relatively minor differences in disposability of the wastes.

4.1.12 Control: No Analysis

The control of the reactors, as it affects the capital and operating costs, or as it affects the nuclear capability, was not considered for the reason that the requirements and costs are extremely nebulous, and that it ~~is, in principle, possible to develop control systems entailing only small~~ nuclear losses. For example, Perry (43) proposes that a batch-loaded, gas-cooled, graphite-moderated breeder reactor be controlled by means of graphite reflectors which are gradually inserted between the core and blanket regions as the fuel is depleted. These reduce the leakage of neutrons from the core and thus maintain the reactors in the critical condition. Similar devices can, in principle, be postulated for the other concepts.

4.1.13 Hazards: No Analysis

The analysis of hazards is difficult and fraught with uncertainty even when applied to systems designed in great detail. It was felt that an attempt to estimate the hazards in the systems studied here, with the exception of the AHBR, would have little meaning in view of the preliminary nature of the conceptual designs.

4.2 Cost Bases

The costs assumed in this study were based, where possible, on current prices and pricing conventions. These are somewhat illusory, especially the price of fuel isotopes and the interest rate charged thereon, and are not expected to prevail when breeders come into common use, which cannot, at the most optimistic, be sooner than 10 to 15 years from now. Nevertheless, it did not seem possible to establish costs bases appreciably more satisfactory to all groups than those presently in use; further, the current bases have the merit that they permit direct comparison with costs predicted for current reactor concepts.

The cost breakdowns have been presented in such a way that other unit costs and rates may be easily substituted for those used.

4.2.1 Value of Fuel Isotopes: \$15/gram

The value of highly enriched U-235 is commonly taken as \$17/gram. While not directly applicable to the case in hand, a study by Jaye, et al (93) of the value of U-233 in five thermal converter reactors utilizing U-235 and thorium gave relative values of the U-233 produced ranging from about \$20 per gram for clean U-233 down to \$10 per gram for material recycled many times. This decline resulted from the ingrowth of U-236, a non-fertile, non-fissile isotope. The current buy-price offered by the U. S. AEC for U-233 contaminated with U-232 is \$15/gram.

The reactors considered here produce a mixture of isotopes having a composition, typically, of 60% U-233, 25% U-234, 7% U-235, and the rest U-236. It does not appear that this mixture could be separated economically into its component isotopes. Some value might be assigned to U-234, as a fertile material, but this would depend upon the value assigned to U-235. It was decided to assign a nominal value of \$15/gram to U-233 and U-235, regardless of composition, and to neglect the influence of U-234 and U-236 on the fissile material value.

During the build-up of the Pa-233 inventory, production of U-233 is decreased by an equivalent amount. Hence, a value of \$15/gram was also assigned to the Pa.

4.2.2 Value of Th-232: \$22/kg (as ThO₂)

Reactor grade ThO₂ was valued at \$22/kg of Th (\$10/lb). Before use, in the concepts studied, ThO₂ must be pelleted, slurried in Bi, or converted to ThF₄ and dissolved in molten fluoride salt. A uniform, nominal charge of \$5/kg of Th was assessed for these operations.

4.2.3 Annual Charges

The annual charge on capital invested in fuel isotopes was taken at 4%; in other materials at 12.7%, which includes interest at 6%, income taxes at 4.6%, and local taxes at 2.1%. Materials not replaced on a regular schedule were amortized by means of a sinking fund at the rate of 1.3% per year, which replaces the material at the end of 30 years. Annual charges on the capital invested in the chemical processing plants was 29% (14% for interest and taxes and 15% for maintenance and operation).

~~4.2.4 Value of Other Materials~~

<u>Item</u>	<u>\$/lb</u>	<u>\$/kg</u>
Reactor Grade Graphite	2.00	4.40
D ₂ O (0.75% H ₂ O)	28.00	61.60
Bismuth	2.25	4.95
LiF (99.99% Li-7)	20.00	44.00
BeF ₂	7.00	15.40

In the MSBR, the fuel carrier salt consisted 63% of LiF and 37% BeF₂. A charge of \$5.00 per lb was made for preparation of the mixture in reactor grade (stripping oxygen, water, HF, F₂), resulting in a final cost of \$40.30/kg. Similarly, the fertile stream carrier salt (67 LiF-18 BeF₂) was valued at \$45.60/kg (excluding ThF₄).

4.2.5 Nuclear Materials Conventions

A convention for assessing use and replacement charges against the fuel and fertile stream carriers was adopted. Thus, in the MSBR and the LBBR, charges were made for the salt and bismuth, but not for the graphite moderator. In the GGBR, the graphite in the fuel plates was charged against

the fuel cycle cost, but the graphite in the reusable cases and the permanent structure was not. In the AHBR, the heavy water used to fill the blanket and cool the fertile material was included in the D_2O inventory, and amounted to more than half the total. Although not strictly a "carrier" for the fertile stream, it was included because it is customary to do so. The same convention was adopted in regard to the DGBR, where all the D_2O in the system was charged, even though it is neither a fuel or fertile stream carrier. Also, in the DGBR, charges were made for the graphite associated with the fuel in the fuel plates.

No charge against fuel cycle cost was made for materials of construction, such as Zircaloy, INOR-8, tantalum, etc.

4.2.6 Chemical Processing Costs

The fuel and fertile stream processes are described in detail in the appropriate subsections in Section 5, and also in Appendix C. The costs of these various processes were studied by Burch, et al (17), and by Guthrie (49).

AHBR. - The fertile stream consists of pellets of ThO_2 . After exposure to neutrons these are ground, dissolved in nitric acid, and treated in a Thorex plant to separate the uranium, thorium, and fission products. The fuel stream consists of a very small stream of oxides of uranium, corrosion products, and rare earths. This is taken up in nitric acid and held for 90 days to allow radioactivity to decay, after which it is combined with the fertile stream for processing in the Thorex plant.

The processing costs for the aqueous reactor were based on estimates prepared by Guthrie (49), which are a revision of estimates made by Ferguson (7).

Table 4.A Capital Investment and Unit Charges for

On-Site Thorex Plants

<u>Capacity</u> <u>kg Th/day</u>	<u>Cap. Inv.</u> <u>\$x10⁻⁶</u>	<u>Ann. Rate</u> <u>%</u>	<u>Ann. Chg.</u> <u>\$x10⁻⁶</u>	<u>Daily Chg.</u> <u>\$</u>	<u>Unit Cost</u> <u>\$/kg Th</u>	<u>Reference</u>
200	11.8	28.5	3.24	10,800	54	Ferguson (7)
1000	20.0	30	6	20,000	20	"
4000	46.4	32.4	15	20,000	12.5	"
600	20.6	(22.1)	4.6	15,300	25.5	Fed. Reg. 1957
300	14.8	29	4.3	14,300	47.7	Task Force (25)

By plotting three unit costs vs capacity, Ferguson obtained the lower curve shown in Fig. 4.2.2. Interpolation at a capacity of 700 kg per day gives a capital cost of \$16.8 million. This is to be compared to a cost of \$20.6 million cited in Federal Register Notice of March 8, 1957, as follows: "The estimated installed cost of the assumed plant, upon which firm daily processing charges will be based, is \$20,570,000. The AEC has determined

Fig.4.2.1 On-Site Thorex Processing
Plant Investment Vs. Capacity. Uranium Product
as $\text{UO}_2(\text{NO}_3)_2$. Cost Includes Dissolution, Decontamination
and Reconversion to ThO_2 Pellets.

that the total cost for the operation of this assumed plant, including waste storage, amortization and allowance for AEC overhead and other intangible expenses is \$4,592,000. Based on this estimated annual cost, a daily cost of \$15,300 (based on 300 days of operation per year) will be the basis for the charge for these fuels which can be processed in the assumed plant as presently conceived." Although not stated in the notice, these figures are for a multi-purpose plant having a capability of treating 1000 kg of Th per day for recovery of U-233, or of recovering U-233 and decontaminating 600 kg of thorium per day (145). The fixed and operating charges amounted to 22% of the capital cost per year, giving a unit cost of \$25.50/kg Th. The reference study (145) was based on construction and operating costs of the pilot plant at ORNL.

The authors of the report of the Fluid Fuel Reactors Task Force (25) cite an estimated cost of \$14.8 million for a Thorex plant having a capacity of 300 kg per day. This estimate was later used by Sargent and Lundy in a report entitled ~~"Power Cost Normalization Studies, Civilian Power Reactor Program-1959"~~ (131). The same capital cost was cited by the AEC in "Status Report on Aqueous Homogeneous Reactors as of 1959" (146) for a plant having a capacity of 500 kg thorium per day. The basis for the increase in capacity was not stated.

The various capital costs estimates are listed in Table 4.A and are plotted in Fig. 4.2.1 vs plant capacity.

Guthrie (49) adopted Ferguson's estimates as a basis, and increased them 20% to allow one recycle of fission products for recovery of U-233 formed by the decay of Pa-233, and for excess cost, if any, of pelletizing ThO₂ over cost of reconstitution of slurry. Guthrie's estimate was used in this study for all concepts utilizing ThO₂ pellets for fertile material. Corresponding unit costs, based on fixed plus operating charges of 29% and 300 operating days per year, are shown in Fig. 4.2.2.

Guthrie also assumed the small stream of spent fuel from the cores of the reactors could be combined, after a holdup of 90 days, with the fertile stream, with no appreciable increase in capital cost of the latter.

Fig. 4.2.2 Reprocessing Costs for On-Site Thorex Plant.

MSBR. - In this concept, both fuel and fertile streams consist of molten salt solutions. The fuel solution contains no thorium, and is treated first with fluorine to remove isotopes of uranium as the hexafluoride, and then is dissolved in concentrated HF containing 10% or less of water. The rare earths and certain other fission products are insoluble, and are separated by clarification or filtration. Soluble fission products are removed by discarding a portion of the salt. The HF is evaporated, the residual salt melted and recombined with UF_4 prepared by reducing UF_6 with hydrogen.

The fertile stream is treated at a rapid rate with fluorine to remove U-233 almost as rapidly as it is formed by the decay of Pa-233. By thus keeping the concentration of fissionable isotopes in the fertile stream low, fissions there are suppressed, and the accumulation of fission products can be controlled by discarding the salt on a 20 year cycle.

The processing costs for molten fluoride salts were studied by Weinrich and Associates (31), who prepared functional flow sheets, sized the equipment, and estimated the costs. For an on-site, continuous plant capable of treating 20 ft^3 of spent fuel per day by both the "volatility" and the "HF dissolution" processes, they estimated a cost of about \$5 million. These detailed estimates were revised at ORNL by W. G. Stockdale to conform with ORNL pricing procedures and reflect ORNL experience with plants for processing nuclear reactor fuels. The revised estimate came to almost \$10 million. This revised estimate was adopted for the present study, and the investment at other capacities was estimated by scaling in proportion to the 0.6 power of the capacity, except that it was assumed that at capacities of $7\text{ ft}^3/\text{day}$ (corresponding to 75-day cycle time) or less, a batch-plant having a cost not less than \$5 million would be used. This lower limit allowed \$1.6 million over Weinrich's estimate of \$3.4 million for a batch plant.

The cost for processing the fertile stream was estimated by extracting cost items for the volatility step, including allocations for instruments, buildings, etc., from the revised estimate for the fuel stream processing. The capital cost thus estimated came to about \$1.8 million for a plant having a capacity of $20\text{ ft}^3/\text{day}$. This was scaled by the 0.6 power of the processing rate.

LBBR. - The fertile stream consists of a slurry of ThO_2 in bismuth. The oxide is separated from the bismuth by flotation with argon in a special head-end process. The capital cost of the equipment, etc. for this head-end process was estimated by Thomas, et al (22) to be \$1.7 million, and this estimate was used for all capacities in this study. The remainder of the fertile stream processing plant is similar to that described for the AHBR, and the cost estimates of Guthrie (49) given in Fig. 4.2.2 were used. This estimate includes an allowance of 20% excess over the cost of a conventional plant for recycling the fission products once for recovery of U-233 formed by the decay of Pa-233.

The capital charges for the fuel-stream processing plant were estimated by Thomas, et al (22) to be \$2.5 million for a plant capable of treating the fuel on a 12 day cycle.

A consideration of the details of the plant shows that the components are small in size and capacity, and leads to the conclusion that the capital cost will be insensitive to changes in capacity. If this is true, there is little to be gained by increasing the processing cycle time, as the unit cost, in mills/kwhr, will remain the same, while neutron losses to fission products, etc., will increase, thus decreasing the breeding credit and the fuel yield. It was concluded that Thomas, et al, have rather well optimized the fuel-stream processing rate, and therefore the rate, and capital costs, were not varied in this study.

GGBR. - The fertile stream in the GGBR consists of pellets of ThO_2 , and handling and processing are similar to that in the AHBR. Accordingly, the estimates of Guthrie given in Fig. 4.2.2 were used.

In the GGBR, the fuel consists of a dispersion of UO_2 in graphite fuel plates. After exposure, these are disintegrated and leached with fuming nitric acid, and the graphite is separated by filtration or clarification. The solution is then processed in the Thorex plant used to treat the fertile stream. The product stream from the Thorex plant consists of an aqueous solution of uranyl nitrate. The fuel is reconstituted by impregnating fresh fuel plates with this solution, after which they are denitrated by firing at 800°C .

The cost of these operations is difficult to estimate, because they are presently only in the laboratory stage of development. The operations must be done remotely because of the activity associated with the presence of U-232 in the recycle stream. A "guesstimate" of the cost was prepared by Guthrie (49), based on a fabrication cost of \$1.5/gram of fissionable isotope in a plant having a capacity of 5 kg/day, which is about the same as current (1960) costs for the MTR and the APPR. Guthrie states "Beryllia or prefabricated graphite shapes would be extra. To achieve these costs, great technical advances would have to be made in fuel element design and fabrication technology. Based on current technology, the investment might be of the order of ten times that shown in" Fig. 4.2.3.

The costs given in Fig. 4.2.3 were used in this study for processing the fuel in the GGBR. An additional allowance at the rate of \$2 per pound for replacement of the graphite fuel plates was made.

DGBR. - In this concept, the fuel and fertile streams are similar to those in the GGBR; accordingly, the same cost estimates were used. Although the fuel element is less rugged, and cylindrical fuel plates would probably be more expensive than flat plates, no additional cost allowance was made.

Fig. 4.2.3. Estimated Investment in Equipment for Remote Prefabrication of Uranium-Graphite Fuel Plates.

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5. DESCRIPTION OF REACTOR CONCEPTS

5.1 Aqueous Homogeneous Breeder Reactor (AHBR)

5.1.1 Introduction

Aqueous homogeneous reactors have been studied extensively at ORNL and a large body of information, including experimental data on the operation of the Homogeneous Reactor Test (HRT) is available (10, 132). A comprehensive report was issued in July 1959 by the staff of the Homogeneous Reactor Project (1). Extensive fuel cycle cost studies were performed by Rosenthal, et al (2, 3, 4). These, and other studies were summarized by Kasten (5). Capital, operating, and maintenance costs of chemical processing plants for aqueous reactor systems have been studied by Ferguson (7), Guthrie, et al (6), and Burch, et al (8).

The reactors employ uranyl sulfate dissolved in heavy water, which serves both as a heat transfer medium and as moderator for the nuclear chain reaction. In both aspects, heavy water is an outstanding material. Also, under proper conditions, many fission products are insoluble in the fuel solution, and may be removed by means of small cyclone separators. The solution is readily contained in an alloy of zirconium, Zircaloy-2, which has a neutron capture cross section satisfactorily low. Difficulty in maintaining the integrity of the Zircaloy core vessel separating the core from the blanket region in the HRT at Oak Ridge has been encountered. The development of holes was attributed to the deposition of solids containing uranium on the core vessel walls during operation at power. Through modification of the core vessel design and method of operation, described in Sec. 5.1.6, this deposition has apparently been prevented in recent experiments on the HRT (148).

The reactor structure is very simple. It consists merely of a cylindrical core region contained in a thin-walled Zircaloy vessel. This is surrounded successively by a blanket region, and a pressure vessel. The fuel stream, under a pressure of 2000 psi, flows through the core where it is heated by the nuclear reaction to about 550°F. The fuel passes thence to externally located heat exchangers, from where it is pumped back to the reactor.

As a result of conferences with the persons listed above, and others, a concept of the most promising system was evolved. The most important difference between the system adopted and those previously studied was in the use of thorium oxide pebbles in the blanket, as proposed by Spiewak and Peebles (9*), rather than a slurry. As will be shown in more detail, a pellet blanket has advantages in respect to uranium inventory and fission product poisoning. Also, the uncertainty in development is thought to be less.

Since all thorium is in the blanket region, it is necessary, in order to obtain breeding ratios in excess of unity, for about half of all neutrons to leak into the blanket. This puts an upper limit of 4 to 5 feet (13) on at least one dimension of the reactor. Because of the high pressure at which aqueous reactors operate, the pressure vessels must be either spherical, or cylindrical with dished ends. Since power density is limited, a large number of spherical reactors would be required in order to produce the power needed. The number of reactors can be reduced by use of cylindrical cores; however, the length-to-diameter ratio is limited by several factors. The most important of these are the requirements that the inlet-outlet nozzle shall be small in diameter compared to the core (area ratio not greater than 1 to ³/₄), and that the velocity of fuel therein shall not exceed ⁴⁰/₂₀ fps in order to limit corrosion and erosion. A length-to-diameter ratio of 3-4 was adopted. The end-leakage from the core, though appreciable, is not large, and the presence of the inlet-outlet nozzle at one end, and of pebble transfer equipment at the other do not impair the performance of the system seriously.

The decision to use thorium oxide as the fertile material was made for the reason that it is the only material whose behavior in water environment is reasonably well known and for which means of chemical processing (Thorex) is well developed and costed. When it was found that the processing of the fertile stream contributed to more than half of the fuel cycle cost, processes based on alternate fertile materials were studied briefly.

* ORNL Internal Report.

5.1.2 AHBR Reactor Concept

The reactor, pump, and heat exchanger are shown schematically in Fig. 5.1.1. The core vessel, composed of Zircaloy plate initially 0.30 in. thick and strengthened by means of ribs and stiffeners, is nominally 4 ft in diameter and 15 ft long. As discussed in Sec. 6.1.1, both of these dimensions are subject to modest adjustment - of the diameter to achieve criticality, and of the length to control power density at the wall of the core vessel. The rate of corrosion was estimated to be ~ 16 mils/year when the power density adjacent to the wall is 50 watts/cc (17). With this corrosion, the minimum wall thickness would be 0.14 in. after ten years, sufficient to withstand a buckling pressure difference of 50 psi. The ten-year-mean equivalent thickness, including ribs and stiffeners, was estimated to be 0.3 in.

The core is surrounded by a blanket region nominally 2 ft thick on the sides, and consisting of three equal-volume annuli of thorium oxide pellets and four annuli of D_2O . The pellets are retained between cylinders of Zircaloy-2, concentric with the core vessel, having a wall thickness of 1/8 in. and perforated by small holes (1/16 in.) to a porosity of 50%. The pellets are cooled by D_2O which enters at the bottom of the reactor vessel at $482^\circ F$, flows upward along the core vessel (thus cooling it), and then percolates through the three layers of pellets (40% voids). It is collected in the outer annulus at $522^\circ F$ and flows to the bottom of the vessel, whence it passes through a pump and heat exchanger (not shown in Fig. 5.1.1) before returning to the reactor vessel inlet. As much as 35% of the total reactor heat may be removed by the blanket cooling system.

Each pellet layer is divided circumferentially into 33 sectors by means of Zircaloy plates in order that distinct batches of pellets may be present in the blanket at all times for reasons discussed in Sec. 5.1.4.

The reactor vessel is about 8 ft in diameter and is constructed of carbon-steel 7 in. thick clad with 0.4 in. of 347 stainless steel, with a design pressure of 2000 psig. The diameter can be made a foot or two larger if it becomes necessary to insert a thermal shield between the blanket and the reactor vessel wall. The length of the vessel is about 25 ft.

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Fig 511 Aqueous Homogeneous Breeder Reactor

Reactor plant data for a typical case are given in Table 5.1.1.

5.1.3 AHBR Fuel System

The fuel consists of uranyl sulfate dissolved in D_2O together with excess D_2SO_4 as required for stability (10, p. 87), fission and corrosion products (partially in solid form), and dissolved D_2 and O_2 . Copper sulfate is not used as a catalyst for internal recombination of radiolytic gases because of the concomitant loss of neutrons to the copper. Instead, a flame recombiner is used by means of which a small amount of superheating may be effected (Sec. 5.1.5). It was assumed that gas production in the core of the reactor will not lead to instabilities. This assumption has been confirmed on a small but not large scale reactor.

Fuel enters the core at $473^{\circ}F$ ($245^{\circ}C$) through an annular nozzle fitted with vanes that impart to it a swirling motion. The swirl stabilizes the flow and prevents boundary-layer separation as the fuel solution flows down along the core-vessel wall. The fuel returns along the axis of the core vessel and exits at $554^{\circ}F$ ($290^{\circ}C$) through a central nozzle leading to the pump, which discharges into two heat exchangers in parallel, whence the fuel returns to the reactor.

The inventory of fissionable material tied up in that portion of the fuel circuit outside the core vessel has an important bearing on the fuel yield of the system. It is desirable to keep this inventory as small as possible, and this is done by making the external portion of the circuit compact. The diameter of the connecting piping is governed by the flow rate and the maximum allowable velocity (20 fps). The length is governed by the requirements for accessibility of reactor, pump, and exchanger for maintenance and replacement. The volume of fuel in the heat exchanger should be minimized by proper selection of tube-wall thickness, diameter, pitch, and length. The various possibilities were thoroughly explored by Spiewak and Parsly (11), who concluded that it would be possible to achieve a power density in the external circuit of about 1.50 Mwt/ft^3 (53.0 kwt/liter). An extract from their study is presented in Appendix A.

In the course of numerous HRT experiments conducted at ORNL (12), it has been observed that there is a permanent loss of fuel from the solution. Presumably, this fuel is absorbed in scale resulting from the deposition of solid products of fission or corrosion (14) on metal surfaces. Since the

Table 5.1.1. AHBR Typical Reactor Plant Data

General

Station net electrical power, Mwe	1000
Station pumping power, Mwe	25.7
Station plant auxiliaries, Mwe	10.4
Station net efficiency, %	27.5
Station thermal power, Mwt	3640
Reactor thermal power, Mwt	910
Number of reactors per station	4
Core geometry	Cylinder
Core diameter and length, ft	4.4 x 14.6
Core volume, ft ³	218
Core moderator material and density, g/cc	D ₂ O; 0.837
Per cent of core occupied by moderator	100
Core vessel material and density, g/cc	Zr; 6.43
Core vessel thickness, in. (10 yr equivalent-mean)	0.30
Blanket thickness, ft	2.0
Blanket volume, ft ³	660
Blanket moderator material and density, g/cc	D ₂ O; 0.857
Per cent of blanket occupied by moderator	78.6
Reflector thickness, cm	5.5
Reflector material and mean density, g/cc	D₂O; 0.857
Thermal shield thickness	None
System operating pressure, psi	2000
Pressure vessel inside diameter, ft	8.8
Pressure vessel thickness, in.	7
Pressure vessel material and clad	Carbon steel, stainless steel
Pressure vessel design temperature, °F	600

Nuclear

Thermal flux, neutron-cm/cm ³ sec	2.5 x 10 ^{14.5}
Per cent thermal fissions	80 - 95
Per cent fissions in fertile stream	10 - 35

Table 5.1.1 (Continued)

Fuel Stream

Character	Fluid
Fuel form	UO ₂ SO ₄
Maximum allowable velocity, ft/sec	20 (in SS)
Per cent of core volume occupied by fuel stream	100
Per cent of end-blanket volume occupied by fuel stream	14
Carrier mean density, lb/ft ³	52
Fissile atom stream-concentration, atoms/cc	0.4 x 10 ¹⁹
Mean heat capacity, Btu/lb-°F	1.21
Maximum temperature, °F	554
Minimum temperature, °F	473
Pressure, psi	2000
Station flow rate, ft ³ /sec	740
Per cent of reactor heat removed from fuel-stream	60-90
Nozzle arrangement in end-blankets	Concentric
Diameter of inlet nozzle, ft	1.9
Stream velocity in end-blankets, ft/sec	20 40 (in zr-u)
Stream velocity in heat exchanger tubes, ft/sec	15
Mean power density in active core, kwt/liter	120
Power density in external fuel-stream circuit, Mwt/ft ³	1.50
Volume of fuel stream per station in:	
Cores, ft ³ (active)	870
End blankets, ft ³	150
Entrance and exit plenums, etc., ft ³	None
External piping, pumps, heat exchangers, valves, etc., ft ³	1790
Dump tanks and miscellaneous	30
Total volume per station, ft ³	2840

Fertile Stream

Character	Solid
Form	Pellets of ThO ₂
Dimensions	1/8 in.
Mean density, lb/ft ³	600

Table 5.1.1 (Continued)

Fertile Stream (Continued)

Per cent of blanket volume occupied by pellets (with 40% voids)	36.1
Thorium mean stream-concentration, atoms/ccx10 ⁻²⁴ (40% voids)	0.013
Blanket structural materials	Zr

Blanket Cooling

Coolant	D ₂ O
Per cent of blanket volume occupied by coolant	78.6
Mean density, lb/ft ³	54.0
Mean heat capacity, Btu/lb °F	1.16
Maximum temperature, °F	522
Minimum temperature, °F	482
Flow rate, ft ³ /sec	260
Per cent of total heat output removed by blanket coolant	10-30
Velocity in heat exchanger tubes, ft/sec	15
Design pressure, psig	2000

Steam Plant

Pressure, psi	416
Temperature, °F	500
Condenser pressure, in. Hg	1.5
Thermodynamic efficiency	28.5

surface area of the heat exchangers is approximately proportional to the rate at which heat is removed from the fuel solution, the amount of uranium adsorbed will also be roughly proportional to the heat removal rate. It was estimated (14) that about 0.034 kg of uranium would disappear from solution per megawatt of heat removed.

A scheme for processing the fuel solution to remove fission and corrosion products is presented in Fig. 5.1.2. D_2 and O_2 formed by radiolysis of heavy water, together with Xe, Kr, and I_2 , are separated from a side stream of fuel solution and sent to a recombiner where the D_2 and O_2 are reunited. The resulting vapor and the I_2 are condensed and returned to the fuel solution. The Xe and Kr are sent to an off-gas system. It is important to remove these, especially the Xe, for failure to do so would result in a loss in breeding ratio in excess of 0.05 units. The problem of Xe removal has been studied by Burch, et al (15); an extract is given in Appendix B. The xenon poison level in the HRT was measured and found to be 0.01 units, which was consistent with the assumption that 60% of the I_2 present was adsorbed on the walls of the fuel system and the rest was in solution. By way of allowance for improvement of the art, it was assumed that the xenon poison level could be maintained at 0.005 units in the AHBR.

The removal of other fission products has also been studied by Burch, et al (17). An extract from the reference cited is given in Appendix C. About 70% by weight of all fission products (mostly rare earths) are insoluble in uranyl sulfate solutions. These, together with insolubles resulting from corrosion of Zircaloy (16 mils/yr) and stainless steel containing Zr, Fe, and Cr are removed from the fuel solution by means of hydroclones (10, p. 304). A hydroclone cycle time of 15 minutes appears to be adequate to control the concentration of insolubles at a very low level. It was estimated that not more than 0.009 from the breeding gain would be lost to rare earths when the fuel concentration is 2 grams of U-233 per liter (17). The concentration of Fe, Cr, and Zr was taken at 10 ppm for purposes of estimating parasitic neutron captures.

Fig 5.1.2 Fuel Processing for an Aqueous Homogenous Breeder Reactor.

Fission products in solution are mostly alkali metals and alkaline earths, together with small amounts of rare earths and miscellaneous metals. The concentration of these is controlled by removing a portion of the fuel solution periodically, treating with D_2O_2 to precipitate UO_4 , and sending the clarified liquid to a Thorex plant (discussed below). The UO_4 is taken up in D_2SO_4 solution and returned to the fuel circuit. Burch, et al (17) have estimated that the loss in breeding ratio resulting from capture of neutrons in soluble fission products can be held to 0.004 by this means. Nickel and manganese ions, which are soluble, and which result from corrosion of stainless steel, will also be removed by this process. Their equilibrium concentrations were estimated to be 0.00167 and 0.00025 molar respectively, corresponding to a corrosion rate of 0.3 mils/yr and a processing cycle time of 30 days. For the purposes of estimating inventory of fissionable material, a hold-up time for the process of seven days was assigned.

The solids in the underflow from the hydroclones contain about 10% by weight of uranium (17). The stream is held for 90 days to allow radioactivity to decay before it is treated in a Thorex plant (see below). The uranyl nitrate produced is converted to uranyl sulfate. The fuel solution must be prepared and handled remotely because it is contaminated with the radioactive daughters of U-232. A supply of fuel sufficient keep the reactor in operation for 30 days with the processing plant shut down is reserved, in order that production of power may continue uninterruptedly.

5.1.4 AHBR Fertile System

A circulating slurry blanket may be processed by the continuous withdrawal of a small side stream. The processing cost is a function of the rate, expressed in kg of Th per day. This, in turn, is fixed by specifying the amount of thorium in the fertile system and the cycle time. The concentration of U-233 in the circulating stream, expressed in grams per kg of thorium, rises to some steady-state level determined, on the one hand, by the rate of neutron-capture in thorium and, on the other, by the rate of removal of U-233 in the process stream.

Alternatively, the processing may proceed batchwise. At the end of the cycle, the whole system is discharged and processed. The average daily rate is the same, but the mean inventory of U-233 in the blanket is only half as great, and the mean inventory of fission products is only one-sixth as great.

These advantages are offset, however, by several disadvantages. In the first place, the rate of generation of heat in the fertile system fluctuates widely - from zero to 40% of the total rate in extreme cases. In the second place, the inventory saved by use of the batch-loading technique would be exactly compensated by the inventory in discharged material waiting to be processed, since it would be uneconomical to build a processing plant larger than that capable of handling the discharged batch in less time than required to produce it.

By the use of pellets of thorium oxide, rather than a slurry, it is possible, without subdivision of the cooling system, to subdivide the blanket in respect to processing.

~~A simple and adequate method of unitary cooling of multiple batches of~~
ThO₂ pellets was described in Sec. 5.1.2. The blanket contains 33 separate and distinct batches of pellets. With a cycle time of, say, 660 days, an irradiated batch is removed every 20 days, and a fresh batch is added. By keeping the batches separate, the time-mean concentration of U-233 in the fertile material in the blanket is only about one-half the concentration of U-233 in the discharge stream of fertile material. Thus, for a given inventory of U-233 in the batch-loaded blanket, only half as much thorium need be processed as with the continuously-processed slurry blanket. At the same time, because the number of batches is 33, the average amount of U-233 in the blanket fluctuates by only 3% at batch-loading time. The heat production rate, etc., are nearly steady with time, and the blanket cooling system may be designed accordingly. Also, the holdup of material waiting to be processed is small.

It is desirable to shift fertile material frequently from regions adjacent to the core vessel wall, where the neutron population is high, to the outer portions of the blanket, where the population is low in order to reduce the captures in Pa. After the protactinium has decayed to U-233,

the material may be returned to the high-flux region. Maximum savings are obtained when the rate of radial circulation is infinite. However, as shown in Appendix D, only a small penalty is exacted when the cycle time for circulation is three days. It appears feasible to achieve this rate by use of the transfer mechanism described by Spiewak and Peebles (9), who propose to raise pellets to the top of the blanket by hydraulic transport in streams of D_2O . Downflow and discharge is by gravity flow.

In the scheme proposed here, the three concentric zones of pellets in the blanket are each divided circumferentially into 33 sectors, or "baskets". Three baskets, one in each zone, are used for each individual batch of pellets, one-third of a batch being charged to each basket. Daily, the pellets from the innermost basket are removed, mixed thoroughly, and charged to the outermost basket. The pellets displaced from the outermost basket are moved to the intermediate basket, and so on. It is important that the three baskets used for a given batch of pellets not lie on a radius of the reactor, but be distributed about the core in such a manner that the concentration of U-233, when averaged over any radius, shall not be a strong function of the angular direction.

After a period of 100 to 600 days, a batch of pellets is removed from the reactor and sent to the processing plant. For reasons given in Sec. 5.1.1, the Thorex process (10, p. 332) was selected. This process utilizes organic and aqueous solvents in successive liquid-liquid extractions to separate uranium and thorium from each other and from fission products. A pilot plant has been in operation at ORNL for several years (16, 18). No especial difficulty in processing pellets is anticipated. It is believed that a delay of ten days before processing will be sufficient to allow the decay of radioactivity to a level such that the material may be safely handled in the extraction plant. Indications are that the protactinium will appear in the waste along with fission products (16). The waste is held to allow the Pa to decay, and then is reprocessed once to recover the U-233 formed. When the value of U-233 is \$15 per gram, and the interest rate is 4%, the optimum holding time is about 200 days.

The recovered thorium oxide is pelleted and recharged to the reactor. This must be done remotely because the recycled material will be contaminated with the radioactive daughters of Th-228.

The processing loss was taken to be 0.1% of the fissile atoms per pass through the processing plant. This material is lost in part by adsorption in the degraded organic material that collects at the interface between the aqueous and organic phases in the extraction plant, and in part as U-233 and undecayed Pa-233 in the waste containing the various separable products of fission and corrosion. This loss is proportional to the average number of passes made by fuel atoms. Since 10 to 35% of all fissions occur in the fertile stream, the probability that an atom of freshly bred fuel passes through the processing plant ranges from 0.9 down to 0.65. The atoms of Pa-233, amounting to less than 10% of the bred fuel, make a second pass. Finally, an atom of fuel put into the fuel stream has a probability of about 0.1 of making another pass through the processing plant via the underflow from the hydroclones. Thus, the average number of passes an atom of fuel makes through the processing plant ranges from about 0.85 to 1.1. A nominal value of 1.0 was adopted, and a uniform allowance of 0.001 loss in breeding gain was assigned to the processing.

Neutron losses to xenon in the fertile stream was based on the assumption that the xenon would be trapped within the pellets of thorium oxide. Losses to samarium isotopes were similarly computed. Losses to other fission products were based on the same assumption, and calculated from the time-mean concentration and a composite cross section of 50 barns per fission* with a resonance integral of 161 barns per fission.

5.1.5 AHBR Power System

In the AHBR, no intermediate coolant is used. Feedwater is heated and steam is generated in the primary heat exchangers by the fuel stream and the blanket cooling stream. The system chosen was adapted from the John Sevier Plant of the TVA (19). A schematic diagram of the steam cycle is shown in Fig. 5.1.3. Steam is produced at 500°F and 416 psia. There are three stages of feedwater heating and one stage of moisture separation, which limits the maximum moisture content to 11%. The pressure in the condenser is 1.5 in. Hg absolute.

* 2200 m/sec

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Fig. 5.1.3. Power Generation System for Aqueous Homogeneous Breeder Reactor

The gross thermodynamic efficiency is 28.4%, including 0.8% added from superheat recovered in the radiolytic gas recombiner. Approximately 36 Mwe are required for pumping in the steam and reactor plants, and for other general purposes. The station efficiency is thus 27.5%.

5.1.6 Status of AHBR Development

The development of the aqueous homogeneous reactor is well advanced compared to the status of the other systems studied. A comprehensive review as of 1959 was issued recently by the U. S. Atomic Energy Commission (132). A second reactor experiment (HRT) is in progress, and has demonstrated the feasibility of remote maintenance and repair. With the exception noted below, the feasibility of the fuel system is well established. A large effort has been devoted to the development of aqueous slurries of ThO_2 particles suitable for use in the blanket; this is being supplemented by studies of the handling and irradiation of pellets of ThO_2 .

Methods of processing the fuel and fertile streams, except as noted below, have been proved in pilot plant operations (hydroclones and Thorex).

The major development problems in the AHBR concept described here are, in the order of their importance:

1. Guaranteeing the integrity of the core-vessel. As mentioned in Sec. 5.1.1, difficulty in the operation of the HRT was ~~encountered in that holes in the core tank developed which~~ allowed an exchange of fluid between the fuel and fertile streams. Convincing evidence was obtained that at least one of the holes resulted from melting of the Zircaloy wall.

In the HRT, the fuel entered at the bottom of the vessel, and flowed up through a number of screens placed in the inlet diffuser section. This flow arrangement favored the accumulation of solids too large and heavy to be carried away in the exit stream at the top of the core, and too large to fall through the screens at the bottom. Since the operation of the reactor was accompanied by a power dependent loss of uranium from the fuel solution, and

1. (Continued)

since there was independent evidence, in the form of power excursions, of the presence of uranium bearing solids in the core, it was presumed that the most likely mechanism for the formation of the holes in the core tank involved the auto-catalytic deposition of uranium bearing solids on the core vessel. The heat release rate in such accumulations would be very high, and the heat removal rate low. Local boiling would probably set in, resulting in the deposition of more uranium from solution, and leading eventually to burn-out of the metal wall.

In recent experiments (148) on the HRT, most of the screens were removed, and the flow direction was reversed. Power surges of the kind previously observed, and thought to be associated with the passage of uranium containing solids through the core, have not been observed. Neither has there been a detectable loss of uranium from solution at powers up to 5 Mw. It is believed the solution is now stable, and that uranium bearing solids are not depositing on the core vessel walls.

However, in view of the importance of the problem, and the possibility that deposition might occur in the present system at higher powers, work is proceeding on core vessel concepts designed to produce flow patterns and heat transfer characteristics unfavorable to the formation and deposition of solids on the core vessel wall (149). Current designs call for the fuel to enter the core vessel tangentially through two opposed slit nozzles running the length of the cylindrical vessel, and to leave through a polar discharge nozzle at the top (150). Provision is also made for removing any dense phase and sediment through a drain line at the bottom of the vessel.

2. Transporting pellets of ThO_2 inside the reactor vessel. The use of pellets is presently under study at ORNL (9). The experience accumulated in the petroleum and chemical industries in handling particulate solids in both fluid and gaseous streams provides guidance here. Attrition of the particles and the accumulation of fines in various parts of the cooling circuit may present problems.
3. Controlling corrosion of the core vessel. Inpile loop tests (151) provide a basis for predicting the rate of corrosion (estimated 16 mils/yr) of the Zircaloy core tank of the AHBR under the assumed operating conditions. However, this requires an extrapolation beyond the experimental range of conditions, and until experiments are available in the desired range, corrosion of the core vessel must be considered a possible major problem in the AHBR. If the rate is higher than anticipated here, this will require either the replacement of the core tank more frequently than the ten years allowed, or the thickness would have to be greater, resulting in a poorer neutron economy.
4. Remotely operating and maintaining machinery for pelleting ThO_2 .

Some additional work needs to be done on the xenon removal system, and a pilot plant for the peroxide precipitation step in the fuel processing remains to be demonstrated. Also, the behavior of protactinium in the Thorex plant is not known precisely - it may adhere to the walls of the various pieces of equipment. However, unless the core-vessel problem proves intractable, the cost of the remaining development work needed to bring the AHBR concept to fruition should not be great.

5.2 Molten-Salt Breeder Reactor (MSBR)

5.2.1 Introduction

Molten-salt fuels are solutions of UF_4 in mixtures of the fluorides of among others, lithium-7, beryllium, and zirconium. These mixtures melt below 1000°F , have negligible vapor pressure at temperatures of interest, and are sufficiently compatible with nickel-base alloys of the INOR series to assure long life for components at temperatures up to 1300°F . They are remarkably stable in radiation fields, and have excellent heat-transport characteristics. In addition, they exhibit ample solubility for ThF_4 , and therefore the necessity of using a slurry to achieve a fluid fertile stream is avoided.

The neutron-capture cross sections of the alloyed components of INOR are too high to permit the alloy to be used in the neutron-active portion of a breeder reactor. However, the preferred moderator material (graphite) is compatible with the fuel-salt, and may be used as structural material in the active core in direct contact with salt on the one hand and metal on the other.

These properties of the molten-salt systems make possible the conceptual design of two-region, near-thermal breeder reactors having the maximum nuclear capability possible for the combination of graphite with the two salt streams. They also make possible reactors which can operate at high temperature, high specific power, and low pressure.

The development of molten-salt fuels was initiated by the ANP project at ORNL, and continued by the Molten Salt Reactor Project. Progress through 1958 was summarized by Lane, et al (10, p. 567). The work includes the successful execution, in 1954, of a reactor experiment (the ARE). Recent work has been reviewed by MacPherson (27). An experimental reactor is being constructed at ORNL (36).

Capital, operating, and maintenance costs of chemical processing plants for molten salt reactors were estimated by Burch, et al (8), who based their findings on a design study by Weinrich and Associates (31).

Fuel cycle costs for the MSBR were also studied by Burch, et al (17), and by Campbell, et al (30, 32, 33).

The molten-salt breeder concept selected for evaluation was a modification of the "bayonet-tube" version described, along with others, by MacPherson (26) in 1959, and evolved through discussions with members of the Molten Salt Reactor Project and other contributing groups at ORNL.

5.2.2 MSBR Reactor Concept

The molten-salt breeder reactor concept examined in this study is pictured schematically in Fig. 5.2.1. The reactor has a cylindrical core $7\frac{2}{3}$ ft in diameter by $7\frac{2}{3}$ ft high surrounded on the sides and ends by a three ft blanket. A graphite reflector one ft thick surrounds the blanket. The fuel stream heat exchanger and fuel pump are mounted directly above the reactor in a vertical configuration in order to minimize the volume of fuel invested in the heat removal circuit and to facilitate maintenance of these components.

The graphite moderator is formed by assembling 7.5 in. square prisms about 7 ft long. The corners of adjacent prisms are machined to form vertical passages of circular cross section about 5 in. in diameter. The fuel salt flows in two pass bayonet tubes which are inserted into these vertical passages; the tubes are made of impermeable graphite. The outer tubes (see Fig. 5.2.1) have inside diameters of 3.75 in. and walls 0.5 in. thick. They are brazed to an INOR-8 metal header (34, 35, 36). These joints are presumed to be substantially leakproof; however, a small leak of fertile stream into the fuel stream could be tolerated for the reason that thorium would quickly be removed in the fuel reprocessing step along with rare earth fission products.

The inner tubes have inside diameters of 2.4 in. and walls 0.25 in. thick. They are joined to the inner plenum of the metal header by slip joints; these joints need not be leakproof since some bypass leakage of fuel at this point can be tolerated. The reactor contains approximately 90 bayonet tubes. Sufficient clearance between the fuel tubes and the graphite moderator is provided to allow for differential expansion between the moderator and the metallic fuel plenum.

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Fig.5.2.1. Molten Salt Breeder Reactor.

The graphite moderator is supported from below and on the sides by suitable graphite members. The regions between the core and the reactor vessel and the annuli between the fuel tubes and the moderator are filled with blanket fluid which is maintained at a pressure higher than that of the fuel fluid. Thus, if a leak develops, fuel will not enter the fertile stream; instead, fertile fluid will enter the fuel stream, and reactivity will decrease.

The reactor vessel, heat exchanger, pump impeller, etc., are constructed of INOR-8. Additional data on the reactor plant are given in Table 5.2.1.

5.2.3 MSBR Fuel System

The fuel salt consists of a mixture of 63 mole % LiF (99.996 % Li-7) and 37 mole % BeF_2 . It contains sufficient UF_4 (equilibrium mixture of uranium isotopes) to make the system critical (about 0.35 mole %), and occupies 16% of the volume of the core. Recent work (135) indicates that 5 mole % ZrF_4 (replacing Be) should be added to reduce the sensitivity of the fuel toward precipitation of UO_2 should the system be contaminated with small amounts of oxygen.

Fuel salt (50,000 gpm) enters the core from an inlet plenum at 1125°F , passes down through the annulus in the bayonet tubes at 20 fps (maximum), rises through the inner tube, and is discharged at 1300°F . It is collected in a metallic plenum and passes up through a duct to the impeller chamber of the pump, from whence it is forced through the tubes of the heat exchanger, and back into the reactor inlet plenum. Changes in volume of the fuel are accommodated in the space above the impeller.

Spiewak and Parsly (11) have estimated the volume of fuel required to fill the portion of the fuel circuit lying outside the active nuclear zone. An extract from their work is given in Appendix A. They concluded that it should be possible to achieve power densities in the external circuit of 7.6 Mwt/ft^3 .

A flow diagram of the processing scheme for the MSBR is given in Fig. 5.2.2. Volatile fission products (mainly xenon and krypton) are removed by sparging the fuel salt in the surge-tank above the pump impeller

Table 5.2.1. MSBR Typical Reactor Plant Data

General

Station net electrical power, Mwe	1000
Station pumping power, Mwe	20
Station plant auxiliaries, Mwe	10
Station net efficiency, %	42.3
Station thermal power, Mwt	2364
Reactor thermal power, Mwt	1182
Number of reactors per station	2
Core geometry	Cylindrical
Core diameter and length, ft	7-2/3 x 7-2/3
Core volume, ft ³	358
Core moderator material and density, g/cc	Graphite; 1.8
Per cent of core occupied by moderator	~ 77
Core vessel	None
Blanket thickness, ft	3
Blanket volume, ft ³	1644
Blanket moderator material and density, g/cc	Graphite; 1.8
Per cent of blanket occupied by moderator	10
Reflector thickness, ft	1
Reflector material and mean density, g/cc	Graphite; 1.8
Thermal shield	None
System operating pressure, psi	100
Reactor vessel inside diameter, ft	~ 16
Reactor vessel thickness, in.	2
Reactor vessel material	INOR-8
Reactor vessel maximum temperature, °F	1300

Nuclear

Thermal flux, neutron-cm/cm ³ sec	2.0 x 10 ¹⁵
Per cent thermal fissions	~ 70
Per cent fissions in fertile stream	2

Table 5.2.1. (Continued)

Fuel Stream

Character	Fluid
Fuel form	UF ₄
Carrier	63 LiF; 37 BeF ₂
Maximum allowable velocity, fps	40
Per cent of core volume occupied by fuel stream	16
Per cent of end-blanket volume occupied by fuel stream	2.67
Carrier mean density, lb/ft ³	119.5
Fissile atom stream-concentration, atoms/cc x 10 ⁻²⁴	0.9 x 10 ²⁰
Mean heat capacity, Btu/lb-°F	0.544
Maximum temperature, °F	1300
Minimum temperature, °F	1125
Liquidus temperature, °F	849
Pressure, psi	100
Station flow rate, ft ³ /sec	178
Per cent of reactor heat removed from fuel stream	91
Fuel-stream channel arrangement in core	Bayonet tubes
Dimensions of fuel-stream channels, in.	3.75
Lattice arrangement and pitch, in.	Rectangular; 7.5
Average stream velocity in core, ft/sec	24
Average stream velocity in heat exchanger tubes, ft/sec	20
Power density in active core, kwt/liter of fuel	740
Power density in external fuel-stream circuit, Mwt/ft ³	7.6
Volume of fuel stream per station in:	
Cores, ft ³	113
End blankets, ft ³	88.4
Entrance and exit plenums, etc., ft ³	
External piping, pumps, heat exchangers, valves, etc.	280.6 ft ³
Dump tanks and miscellaneous, ft ³	48.2
Total volume per station, ft ³	530.2

Fertile Stream

Character	Fluid
Thorium form	ThF ₄
Carrier	67 LiF; 18 BeF ₂ ; 15 ThF ₄
Per cent of core volume occupied by carrier	7

Table 5.2.1. (Continued)

Fertile Stream (Continued)

Per cent of upper end-blanket occupied by carrier	74
Thorium stream concentration, atoms/cc	0.4×10^{22}
Carrier mean density, lb/ft ³	191.7 (includes ThF ₄)
Mean heat capacity, Btu/lb °F	0.32
Maximum temperature, °F	1300
Minimum temperature, °F	1150
Pressure, psi	100
Liquidus temperature, °F	932
Station flow rate, ft ³ /sec	22.6
Per cent of reactor heat removed from fertile stream	10
Fertile-stream channel arrangement in core	Annuli around fuel tubes
Width of fertile-stream annuli in core, in.	~ 0.25
Stream velocity in heat exchanger tubes, ft/sec	14

Volume of fertile stream per station in:

Cores, ft ³	100
End and side blankets, ft ³	2880
Entrance and exit plenums, etc., ft ³	-
External piping, etc.	420
Dump and surge tanks, ft³	2600
Total volume per station, ft ³	6000

Steam Plant

Pressure, psi	1800
Temperature, °F	1050
Condenser pressure, in. Hg	1.5
Thermodynamic efficiency	43.5

FIGURE: 5.2.2 FUEL PROCESSING CYCLES FOR MOLTEN SALT BREEDER REACTOR

with helium. The off-gas is passed through charcoal beds where xenon and other volatile fission products are adsorbed. Helium is recovered for re-use. It is believed that neutron losses to xenon, etc., can be maintained at 0.005 neutrons per neutron absorbed in atoms by this means (8).

The fuel-salt is treated to remove non-volatile fission products by a combination of the fluoride volatility process (10, p. 591, and 27, p. 17) and the HF-dissolution process (32). The fuel salt is first treated with fluorine. Uranium tetrafluoride is converted to volatile UF_6 and swept out of the fluorinator. The UF_6 is reduced to UF_4 with H_2 in a flame-reducer (37), and recycled to the reactor. Uranium losses in the processing cycle were taken as equivalent to 0.0022 neutrons per neutron absorbed in fuel regardless of the processing rate, and the losses were assumed to occur entirely in the $UF_6 \rightarrow UF_4$ reduction step. This assumption is reasonable because the fluorination of UF_4 goes very rapidly and to completion. Plant experience (61) in the reduction of UF_6 has shown that the unit process can be carried out with almost arbitrarily small process losses. In fact losses are customarily smaller than random errors in sampling and analysis. Uranium losses no larger than 0.01% of throughput are considered to be attainable with a reasonable amount of operational care. The loss assumed is median between those assumed for the AHBR (~ 0.001) and the LBBR (~ 0.003).

The uranium-free salt, containing fission products, is dissolved in concentrated HF (10% H_2O). Bare earths and Zr ions are insoluble, and are separated by filtration or clarification. The solution, containing recovered LiF and BeF_2 along with soluble fission products (principally Cs, Rb, Sr, Ba, Te, Se, Nb, Cd, Ag, and Tc) is evaporated to dryness. Part of the salt is discarded to rid the system of the soluble fission products, and the rest is melted, combined with recovered uranium and makeup ZrF_4 , and recycled to the reactor.

The sum of the parasitic captures (neutrons per neutron absorbed by fissile atoms) in the two groups of non-volatile fission products is designated the "poison fraction". A given poison fraction may be obtained by a several different combinations of cycle times for the HF dissolution process and the discard step, and for a given poison fraction there is an

optimum combination of cycle times that results in a minimum total cost. This optimum combination was determined for every case by the method described in Appendix E. The effective capture cross sections for about 40 isotopes were estimated from the thermal cross sections and the resonance integrals tabulated by Nephew (38). The equilibrium concentration and the resulting poison fraction of each isotope was computed from the cross section, the power, the yield, and the removal rate.

5.2.4 MSBR Fertile System

The fertile stream is composed of LiF , BeF_2 , and ThF_4 in the proportions 67-18-15 mole %. The equilibrium mixture also contains Pa-233, uranium isotopes, and a small amount of fission products. As shown in Fig. 5.2.1, a portion of the fertile stream flows up through the core in the annular spaces between the bayonet tubes and the graphite moderator. About 6% by volume of the core is occupied by the fertile stream, which removes about 10% ($\sim 112 \text{ Mw}$) of the reactor heat. However, only 1-6% of the system energy is generated by fissions in this stream.

The fertile stream processing consists of a fluoride volatility step only. Blanket salt is fluorinated to remove bred uranium as the hexafluoride; salt from the fluorinator is recycled directly to the blanket without further treatment, unless stripping with He or other treatment to remove traces of F_2 is required first. In the MSBR, such a small fraction of the fissions occurs in the fertile stream that the HF dissolution step is unnecessary. To control fission-product poisoning, it is sufficient to replace the fertile-stream carrier on a 20-year cycle. Protactinium is not removed from the fertile stream in this process; its concentration rises until its rate of decay equals its production rate. Loss of neutrons to protactinium is controlled by adjusting the volume at the fertile stream. Larger volumes result in smaller losses but greater inventories of salt, uranium, and thorium.

UF_6 from the fertile-stream fluorinator is reduced to UF_4 with H_2 and blended with the fuel stream for recycle to the reactor. Excess production is sold. Loss of uranium was estimated at 0.1% per pass through the processing cycle.

The concentration of fission products (treated as a single group) in the fertile stream was computed from the yield and the discard rate. No allowance for burnout was made. The poison fraction was then computed from the concentration and a "composite" cross section estimated by Nestor (24) of 50 barns and a resonance integral of 161 barns (Appendix F).

Xenon, etc., is removed by sparging the fertile stream with helium at some convenient point; neutron losses are negligible.

5.2.5 MSBR Power System

The fuel-stream heat exchanger (Fig. 5.2.1) is constructed of INOR-8 and has about 8000 tubes 0.375 in. O.D. with walls 0.028 in. thick. The design and arrangement of both the heat exchanger and the pump are such that the volume of fuel invested is a minimum (11). The fertile stream heat exchanger (not shown in the figure) contains somewhat less than 1000 INOR-8 tubes. The volume of the fertile stream is governed by the need for a large inventory of thorium to control loss of neutrons to protactinium (Sec. 3.2.2).

The secondary coolant in both fuel- and fertile-stream heat exchangers is sodium. This is not compatible with either stream, as contact with sodium will cause the precipitation of, among other things, uranium. If satisfactory heat exchangers cannot be developed, then an intermediate coolant consisting of a barren mixture of lithium and beryllium fluorides can be used (26, 27). Such a mixture would be compatible with both salt and sodium.

The steam cycle for the MSBR was adapted from the John Sevier Plant of the TVA (19). A schematic diagram is shown in Fig. 5.2.3. Steam is produced at 1800 psi and 1050°F. There is one stage of reheat and eight stages of regenerative feedwater heating. The condenser pressure is 1.5 in. Hg absolute, and the thermodynamic efficiency was calculated to be 43.5% (11). Additional data are given in Table 5.2.1.

Fig. 5.2.3. Power Generation System for Molten-Salt Breeder Reactor.

5.2.6 Status of MSBR Development

A molten-salt reactor experiment (the ARE) was successfully performed in 1954. An experimental reactor (the MSRE) is currently under construction at ORNL (39). This reactor is expected to demonstrate the compatibility, under radiation, of the 3 phase system composed of graphite, salt (containing uranium and fission products), and INOR-8. The reliability and maintainability of pumps, heat exchangers, control devices, etc., will also be demonstrated. Breeding, however, will not. The reactor employs but a single fluid, and this will contain only a token amount of thorium, if any.

The fluoride volatility process has been reasonably well demonstrated in a pilot plant that has operated successfully at ORNL for several years (32). The feasibility of the HF dissolution process has not been demonstrated beyond the laboratory stage, and costs are nebulous. The efficiency of removal of the rare earths and related isotopes is not fully known, nor has the behavior of the precipitates in pilot-scale equipment been studied.

The choice of a two-region core for the MSBR presents problems both of design and development. Since there are two different fluid streams in the reactor, there must somewhere be a reliable, reasonably leak-proof, graphite-metal joint having fertile stream on one side and fuel stream on the other. Graphite-graphite joints may also be required. There are severe problems resulting from the difference between the thermal expansion coefficients of INOR and graphite, and from dimensional changes and deterioration of physical properties of graphite in radiation fields. The production of low-permeability graphite in the required sections is thought to be feasible by a modest extension of present technology (26), but has not been demonstrated.

It is important that low-permeability graphite be available for construction of fuel tubes for the MSBR because of the possibility of xenon adsorption and retention. If xenon should be adsorbed to the extent that its only removal mechanism is by burnout, the contribution to the poison fraction would be as high as 0.05 neutrons absorbed in xenon per neutron absorbed in fuel. The method of Watson and Evans (147) was used to estimate the degree of diffusion of xenon into graphite (Appendix B) for several values of the diffusion coefficient. It appears that if the diffusion

coefficient is no greater than 10^{-7} cm²/sec, the xenon poison fraction can be held to 0.005 by sparging the fuel stream on a 55-sec cycle. If the diffusion coefficient is as large as 10^{-5} cm²/sec, the same poison fraction can be attained by sparging on a 27-sec cycle. It is doubtful that a molten salt breeder reactor can attain a breeding ratio greater than unity unless xenon can be removed via a sparging cycle.

Provision for auxiliary heating during startup and shutdown to prevent freezing of the fuel-salt is a disadvantage of the molten-salt systems, but not a major difficulty.

5.3 Liquid Bismuth Breeder Reactor (LBBR)

5.3.1 Introduction

Molten bismuth possesses many advantages as a fuel carrier in fluid fuel reactors. It has an appreciable solubility for metallic uranium, its neutron capture cross section is satisfactorily low, it is immune to radiation damage and thermal decomposition, and its vapor pressure is low at temperatures of interest. In principle, it is possible to continuously extract fission products from the fuel stream and add fresh fuel by liquid-liquid extraction techniques using high-temperature molten salts. Because its thermal conductivity is high, compared to those of non-metallic liquids, heat can be added or removed from bismuth with small temperature drops, but because of its low volumetric heat capacity, large volumes of bismuth must be pumped if the heat removal rate is to be high.

Bismuth containing uranium and fission products is compatible with graphite. It can be contained under isothermal conditions in carbon steel, and in tantalum with temperature gradients.

Unfortunately, thorium is not soluble in bismuth, and in order to use it as the carrier for the fertile stream, it is necessary to employ a slurry of metallic thorium alloys or a compound of thorium. Thorium oxide has nearly the same density as bismuth at temperatures of interest, is chemically compatible, and appears to form a stable suspension.

Liquid bismuth reactors have received extensive study and development at Brookhaven National Laboratory and at Babcock and Wilcox Company in Lynchburg, Virginia (136, 137). A detailed study of liquid bismuth breeders was performed by Thomas, et al (22), for the AEC. The LBBR concept chosen for evaluation was developed by consultation with Thomas and his co-workers, and by reference to their report. In order to simplify the calculations, a certain amount of idealization was necessary, mainly in regard to the method of handling the heterogeneity and geometry of the core in the nuclear calculations. The recommendations of Thomas were followed in regard to efficiency and cost of fuel and fertile stream processing, and in regard to parasitic captures in bismuth. Somewhat greater optimism in regard to the volume of fuel in reactor downcomer was employed.

In the configuration chosen, it is possible to pass part of the fertile stream through the core. This conveniently provides cooling for the graphite moderator and also provides an additional degree of freedom in reconciling the nuclear, thermal, and mechanical requirements of core design.

5.3.2 LBBR Reactor Concept

A schematic diagram of the reactor is shown in Fig. 5.3.1. Neutron moderation is obtained by the use of graphite in the core, blanket, and reflector regions. The graphite matrix forming the structural portion of the reactor is constructed by fitting together and cementing special shapes having premachined flow channels. The use of high-density (1.8 g/cm^3), low-permeability graphite having a microscopic cross section of 3.7 mb is assumed. As shown in the figure, the core, consisting of a right circular cylinder with length-to-diameter ratio of unity, is about $6\text{-}1/2 \text{ ft}$ in diameter. It is surrounded by a 4-ft-thick blanket (50 vol % graphite) on all sides, which in turn, is surrounded by a 1-ft-thick graphite reflector.

The moderator structure is penetrated by two sets of channels - one each for the fuel and fertile streams. The fuel channels extend from one outer edge of the blanket region to the other, as shown in Fig. 5.3.1. The fertile stream is delivered to the fertile stream channels through internal headers and is received in the same. The joints in the moderator structure joints must be reasonably leakproof in order to prevent exchange of fluids between the two streams.

The reactor vessel is constructed of stainless steel (to provide high-temperature strength) 2.1 in. thick clad with 200 mils of carbon steel (~~to provide resistance to corrosion by bismuth~~). In order to reduce non-nuclear volume of fuel, all pumps and heat exchangers are located inside the reactor vessel directly above the neutron-active region. There is an extended graphite-to-metal seal between the bottom of the heat exchanger shroud and the top of the reflector, but this seal need not be leakproof because fuel solution is present on both sides. The seals between heat exchanger inlet and reflector and between fertile stream pump housing and reflector are less extended but need to be leakproof to avoid mutual contamination of the fuel and fertile streams.

The heat exchanger and its shroud are constructed of tantalum which resists mass transport by bismuth even in the presence of large temperature gradients. The bismuth in contact with pump impeller, shroud, and reactor

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Fig. 5.3.1. Liquid Bismuth Breeder Reactor.

vessel is all at the same temperature, and here stainless steel clad with carbon steel is sufficient to provide strength and corrosion resistance.

An arbitrary limit of 35 fps was imposed on the fuel stream velocity in the core and end blankets, and 10 fps elsewhere. Because of the high density of bismuth, pressure drops and dynamic loads become high at velocities above 35 fps.

Typical reactor plant data are given in Table 5.3.1.

5.3.3 LBBR Fuel System

The fuel stream, a solution containing approximately 0.3 atom per cent of uranium metal in bismuth, together with smaller amounts of Mg, Zr, and fission products, flows upward through the bottom-end-blanket, through the core and upper end-blanket, and into a plenum. Leaving the plenum at 1300°F, the fuel passes through a tantalum heat exchanger where its heat is given up to an intermediate (sodium) heat transfer system. The fuel-stream is discharged at 1000°F into a sparging chamber where volatile fission products are stripped by helium. Pumps, taking their suction from the sparging chamber, return the fuel-stream through an annulus lying between the reflector and reactor vessel to the bottom plenum. In a typical case the fuel solution occupies 20 volume percent of the core and a similar percentage of the end-blankets. A square lattice spacing of ~ 4 in. was used for the 2 in. diameter fuel channels.

The fuel-stream volume in the core, end-blankets, and heat exchanger tubes can be calculated with considerable confidence. The volume of the remainder of the system is dependent on layout and philosophy. A study was made by Spiewak and Parsly to evaluate holdup requirements in the external system (11). A power density of 3.35 Mwt/ft³ for the external system (excluding dump tanks) was adopted from their work (Appendix A).

That portion of fuel-system volume lying between the circulating-pump discharge plenum and the core-inlet plenum was estimated by Thomas (22) to amount to about 220 ft³, which is about 20% of the total fuel volume. It was felt this volume could be reduced to 135 ft³ by improved design and layout, e.g., by utilizing an internal return system or a two-pass core.

Table 5.3.1. LBBR Typical Reactor Plant Data

General

Station net electrical power, Mwe	1000
Station pumping power, Mwe	46.5
Station plant auxiliaries, Mwe	10.5
Station net efficiency, %	42.1
Station thermal power, Mwt	2376
Reactor thermal power, Mwt	1188
Number of reactors per station	2
Core geometry	Cylindrical
Core diameter and length, ft	6-1/2x6-1/2
Core volume, ft ³	208
Core moderator material and density, g/cc	Graphite; 1.8
Per cent of core occupied by moderator	72.8
Core vessel	None
Blanket thickness, ft	4.0
Blanket moderator material and density, g/cc	Graphite; 1.8
Blanket volume, ft ³	2150
Per cent of blanket occupied by moderator	50
Reflector thickness, ft	1.0
Reflector material and mean density, g/cc	Graphite; 1.8
Thermal shield	None
System operating pressure, psi	275
Reactor vessel inside diameter, ft	~ 16
Reactor vessel thickness, in.	2.3
Reactor vessel material and clad	St. St., Carb. St.
Reactor vessel design temperature, °F	1100
Reactor vessel design pressure, psig	275
Reactor vessel maximum mechanical stress, psi	12,400
Reactor vessel maximum allowable stress, psi	12,500

Table 5.3.1. (Continued)

Nuclear

Thermal flux, neutron-cm/cm ³ sec	1.2 x 10 ¹⁵
Per cent thermal fissions	75
Per cent fissions in fertile stream	18

Fuel Stream

Character	Liquid
Fuel form	U metal
Carrier	Bismuth
Maximum allowable velocity, fps	35
Per cent of core volume occupied by fuel stream	20
Per cent of end-blanket volume occupied by fuel stream	20
Carrier mean density, lb/ft ³	600
Fissile atom stream-concentration, atoms/cc x 10 ⁻²⁴	0.6 x 10 ²⁰
Mean heat capacity, Btu/lb-°F	0.038
Maximum temperature, °F	1300
Minimum temperature, °F	1000
Liquidus temperature, °F	910*
Pressure, psi	275
Station flow rate, ft ³ /sec	259
Per cent of reactor heat removal from fuel stream	80 - 90
Fuel-stream channel arrangement in core	One-pass
Diameter of fuel-stream channels in core, in.	2
Lattice arrangement and pitch, in.	Square; 4
Average stream velocity, ft/sec	20
Average stream velocity in heat exchanger tubes, ft/sec	10
Power density in active core, kwt/liter of fuel	1010
Power density in external fuel-stream circuit, Mwt/ft ³	3.35
Volume of circulating fuel stream per station in:	
Cores, ft ³	83
End blankets, ft ³	104
Entrance and exit plenums, etc., ft ³	402

* Temperature at which Uranium precipitates.

Table 5.3.1. (Continued)

Volume of circulating fuel stream per station in: (Con't)	
External piping, pumps, heat exchangers, valves, etc., ft ³	559
Dump tanks and miscellaneous, ft ³	115
Total volume per station, ft ³	1262
<u>Fertile Stream</u>	
Character	Fluid
Thorium form	ThO ₂
Carrier	Bismuth
Per cent of core volume occupied by thorium carrier	7.2
Per cent of end-blanket volume occupied by thorium carrier	30
Thorium stream concentration, atoms/cc	0.3 x 10 ²²
Carrier mean density, lb/ft ³	607
Mean heat capacity, Btu/lb °F	0.038
Maximum temperature, °F	1300
Minimum temperature, °F	1000
Melting point, °F	488
Station flow rate, ft ³ /sec	69.2
Per cent of reactor heat removed from fertile stream	10 - 20
Fertile-stream channel geometry in core	Cruciform slots
Dimensions of fertile-stream channels in core, in.	2 x 1/4
Fertile-stream channel geometry in side blankets	Cylindrical
Fertile-stream channel diameter in side blankets, in.	2.75
Stream velocity, ft/sec	10
Stream velocity in heat exchanger tubes, ft/sec	10
Power density in external fertile-stream circuit, Mwt/ft ³	2.54
Volume of fertile stream per station in:	
Cores, ft ³	30
End and side blankets, ft ³	2045
Entrance and exit plenums, etc., ft ³	
External, piping, pumps, heat exchanger, valves, etc., ft ³	200
Dump tanks and miscellaneous, ft ³	
Total volume of fertile stream, ft ³ per station	2275

Table 5.3.1. (Continued)

Steam Plant

Pressure, psi	1800
Temperature, °F	1050
Condenser pressure, in. Hg	1.5
Thermodynamic efficiency, %	42.1

A breakdown of the fuel-volume for a typical case is given in Table 5.3.1. The third item under "volume" of 402 ft^3 is composed of 270 ft^3 per station in the downcomer annulus between the radial reflector and the reactor vessel plus 132 ft^3 in the inlet plenums at the bottoms of the vessels. The fourth item of 559 ft^3 was obtained by multiplying the station heat-rate (2376 Mwt) by the fraction of reactor heat removed from the fuel-stream (0.786) and dividing by the power density in the external circuit (3.35 Mwt/ft^3) as estimated by Spiewak and Parsly (11). Ten percent of the sum of the first four items was then added (fifth item; 115 ft^3) to provide for fuel left as heels in dump tanks and miscellaneous small contributions.

Xenon and other volatile fission products are removed from the circulating fuel stream by sparging with helium. As with the AHBR and MSBR, it was assumed that losses to xenon, etc., could be maintained at 0.005 neutrons per atom of fuel destroyed. The off-gases are passed through charcoal beds where the fission products are adsorbed. The helium is recycled to the reactor.

The fuel stream is purified continuously by the BNL salt-extraction process (22). A schematic diagram of the chemical processing plant is shown in Fig. 5.3.2. This system is described in detail in reference 22; only a brief description is presented here.

The nonvolatile fission products in a bismuth reactor fall into two groups: (a) the FPS group, which includes those elements forming chlorides more stable than UCl_3 , and (b) the FPN group, which includes those elements forming chlorides less stable than UCl_3 . The first group is extracted by a salt-stream. A portion of the circulating fuel-stream is withdrawn and passed through the salt extraction plant where the FPS's are removed in a two-stage extractor by contact with a mixture of the chlorides of uranium, sodium, zirconium, and magnesium. The stream concentrations are regulated so that most of the FPS's and part of the Mg are transferred to the salt stream, and the bismuth stream is enriched with uranium. Uranium is recovered from the salt stream by contact with a second stream of bismuth containing magnesium. The FPS's are then extracted from the salt in another contactor by a stream of molten lead. The FPS-lead mixture is stored for disposal.

Fig. 5.3.2. Processing for a Liquid Bismuth Breeder Reactor.

The FPN's are removed periodically from the fuel stream by passing a batch of fuel through a cold trap where Mo, Ru, Rh, and Te crystallize. Most of the batch is then returned to the reactor system, but a small portion is passed through a zinc slagging process. Here the uranium is first extracted by a salt stream containing just enough BiCl_3 to oxidize the elements in the emf-series down to and including uranium. The remaining fission products are removed by slagging the bismuth with zinc, skimming, and then contacting with a second salt stream containing excess BiCl_3 . Magnesium is added to the bismuth, which is contacted with first salt stream to recover uranium and then returned to the reactor (Appendix C).

Losses to fission-products other than xenon were treated as follows. The FPS poison fraction as a function of exposure (for a mean η of 2.224) was taken from reference 22 and normalized with respect to power level and system volume (see Appendix E). The abscissa in Fig. E.2.1 is the quotient of reactor power level expressed in Mwt and chemical processing rate in ft^3/day . The chemical processing rate was held constant at the value used in reference 22 for all cases. The fuel-cycle time is thus determined by the fuel system volume. To calculate the FPS poison fraction, one corrects values taken from the curve in Fig. E.2.1 for variations in η and the fraction of fissions occurring in the fuel stream.

Fig. E.2.2 (Appendix E), shows the fuel-stream FPN poison fraction as a function of exposure. This was also taken from reference 22 and treated in the manner described above. The total fission-product poisoning (excepting that of gases) in the fuel-stream is the sum of the computed FPS and FPN poison fractions. The advantage of this scheme for assigning the fission-product poisoning is that the chemical plant costs given in reference 22 can be used in the calculation of fuel-cycle costs.

Little data are available on the rate of corrosion of tantalum or carbon steel by bismuth, and no allowance for neutron losses to corrosion products was made.

Although the processing plant would be operated as a continuous-flow plant in order that the equipment might be small and economical, spent fuel would probably be withdrawn from the reactor no oftener than once a day. The withdrawal schedule determines the fuel-processing plant holdup of fuel inventories.

5.3.4 LBFR Fertile System

The fertile stream, consisting of a slurry of thorium oxide particles in bismuth (12 wt % ThO_2), is distributed through an internal annular plenum (Fig. 5.3.1) to the outer portion of the blanket, where it flows downward and is collected in another internal plenum near the bottom of the blanket. The bottom plenum distributes the stream to risers, some of which pass through the core and end-blankets, while the remainder pass through the inner portion of the radial blanket. The discharge from the risers is collected in another internal plenum. Leaving at 1300°F , the stream passes through a tantalum heat exchanger where its heat is given up to an intermediate (sodium) heat-transfer system. Discharged from the heat exchangers at 1000°F , the fuel stream passes to the head-volume above the pumps, where the volatile fission products are sparged, and from there is pumped to a plenum serving downcomers which lead back to the top internal plenum.

In a typical case the fertile stream composes 7.2 volume percent ~~in the core, 30 percent in the end blankets, and 50 percent in the side~~ blanket. The fertile stream channels in the core region have a cruciform shape (intersecting slots 0.025 in. by 2.0 in.) to increase the surface-to-volume ratio over that obtainable with circular flow channels. The diameter of the fertile stream channels in the side and end blankets is 2.75 in.

The materials of construction are graphite (inside the reactor), tantalum for portions in contact with the 1300°F streams, and stainless steel clad with carbon steel for the remainder of the system.

A breakdown of fertile stream volume for a typical case is given in Table 5.3.1. A nominal volume of 100 ft^3 per reactor was allowed in the external system for pumps, piping, heat exchangers, etc., which, in the cases cited correspond to a removal power density of 2.54 Mwt/ft^3 .

The fertile system is processed semi-continuously with cycle times ranging from ~250 to ~500 days. The processing includes a special head-end treatment to separate particles of ThO_2 from the bismuth stream and a modified Thorex plant for recovery of thorium and uranium. The bismuth- ThO_2 slurry is broken by flotation with argon. The foam is mixed with mercury and filtered. The filtrate is held in the filters and cooled with mercury for ten days. Residual mercury is then evaporated in a still and the solids are blown with argon to a cyclone separator. The dried ThO_2 enters the modified Thorex plant (Appendix C), where fission products are removed in the first stage of extraction, and thorium separated from uranium in the second. The protactinium accompanies the fission products, which are held for 200 days and then recycled once to recover uranium. Thorium nitrate from the second stage of extraction is converted to ThO_2 and slurried in bismuth for recycle to the reactor. Uranyl nitrate from the second stage of extraction is converted to uranium chloride and added to the fuel stream in the salt-extraction plant described in the preceding section. The holdup time for this latter operation was estimated to be no more than one day.

The chemical plant loss of 0.032 atoms of fuel per neutron absorbed by fuel, estimated by Thomas et al (22), was adopted for this study. The thorium loss was taken to be 1% per pass through the Thorex plant.

The calculation of parasitic captures by fission products in the fertile stream was based on the fission rate in the fertile stream and the assumption that all fission fragments, including xenon and samarium, would be trapped inside particles of ThO_2 . A thermal cross section of 50 barns and a resonance integral of 161 barns was used for a composite group excluding xenon and samarium (24, see also Appendix F).

5.3.5 LBBR Power System

The power generation system is modeled after the John Sevier plant of the TVA (19) and is identical with that specified for the MSBR in Sec. 5.2.1. A schematic diagram is shown in Fig. 5.2.3. Steam conditions are 1800 psia and 1050°F, and the cycle employs one stage of reheat and eight

stages of regenerative feedwater heating, including one deaerator feedwater heater. The condenser pressure is 1.5 in. Hg absolute. The overall station efficiency was estimated to be 42.1% by Spiewak and Parsly (11).

The gross electrical output of the station was taken as the sum of the net output, the pumping power, and 1% of the net output for other station auxiliaries. The pumping power includes reactor as well as steam plant pumping requirements, and amounts to 4.65% of the net output.

Sodium is employed as an intermediate coolant to transport the heat from the reactor fluids to the steam generators.

5.3.6 Status of LBBR Development

A reactor experiment on bismuth systems comparable to the experiments on aqueous systems (HRE-1 and HRE-2) and on salt systems (ARE and in progress MSRE) has not been performed, nor is one currently scheduled in the USA. In-pile tests of fuel and fertile streams have been made. Most of this work was summarized by Thomas (22, p. 10-1).

Bi-ThO₂ slurries were stable in isothermal, static tests, and in non-isothermal, dynamic tests. It appears to be feasible to manufacture and machine relatively large blocks of graphite having sufficient impermeability with respect to bismuth. The components of the system (graphite, bismuth solutions of uranium, and metal container) appear to be chemically compatible, except that the degree (if any) of carburization of steel and tantalum is not known.

The major development problems in the LBBR appear to be the following:

1. Pilot-plant demonstration of the feasibility of operation, control, and maintenance of the fuel-stream processing, which so far has been carried only through the test-tube stage. Precise control of concentrations in several of the streams appears to be important, and the means of securing this is not clear.
2. Demonstration of the integrity of the graphite structure in the reactor, especially with regard to cracks resulting from stresses produced by differential thermal expansion

and radiation damage. Such cracks would contain stagnant fuel or fertile fluid, and might allow the two fluids to mix.

3. Demonstration that fission products are not adsorbed on or absorbed by graphite in the core. In particular, if the rate of diffusion of xenon into graphite is comparable to the rate at which it is removed from the bismuth by sparging, then an appreciable amount will accumulate in the voids (15 % even in graphite having densities as high as 1.9 g/cc), and result in a serious loss of neutrons (as many as 0.05 per fuel neutron absorbed in fissile atoms).
4. Demonstration of fabricability and performance of an all-tantalum, high-duty, low-volume heat exchanger. The art of working tantalum is little known.

Minor development problems include the demonstration of the head and tail-end processes for the fertile stream processing, the demonstration of high capacity, low holdup pumps for bismuth and bismuth slurries, and demonstration of reliable graphite-metal seals that permit relative motion of the graphite and metal to accomodate differential expansion and radiation-induced volume changes.

5.4 Graphite-Moderated Gas-Cooled Breeder Reactor (GGBR)

5.4.1 Introduction

Graphite possesses advantages as a moderator for solid fuel reactors, since it may serve simultaneously as moderator and structural material and is capable of operating at high temperatures. As permanent structure, it can be used to form channels in which the fuel may be placed and through which the coolant may be passed. Its thermal, mechanical, chemical, and nuclear properties are such that it can be used both as structure in the fuel element and as diluent for the fuel.

Cladding on a fuel has two main functions - to achieve chemical compatibility between the fuel and the coolant, and to contain fission products so that a "dirty" coolant system can be avoided. But the reactor performance, especially in regard to specific power, is impaired when a clad is used. Even the use of zirconium would seriously limit the nuclear capability; however, alloys or ceramics based on beryllium might have acceptable nuclear properties. But any clad tends to reduce the heat flux obtainable from the fuel. Thus, there are many incentives to dispense with a clad, and live with a "dirty" coolant.

In a gas-cooled reactor, high specific power is associated with high pressure because the volumetric heat capacity of gases is low. The pressure must be pushed to the limit, and the limit is determined by a number of conflicting system requirements and material properties. The vessel must be large enough to satisfy the thermal and nuclear requirements, i.e., it must house the core, blanket, reflector, thermal shield, and fuel and fertile element handling machinery, and provide for distribution of coolant over the flow area in the core, and collection of outlet coolant. As shown subsequently, the vessels are larger than 10 ft in diameter, and as much as 40 ft long. At the present time, pressure vessels cannot be fabricated in these large sizes with walls much thicker than 10 to 12 in. Also, vessels having diameters greater than ~ 13 ft cannot be shipped by rail. Based on these considerations, the inside diameter of pressure vessels was limited to about 12 ft.

The cost of such a vessel, if fabricated of exotic, high performance materials, would be prohibitive. The use of ordinary carbon steel, however, enables pressures up to 2000 psi to be used.

The high power densities possible with gas-cooled, graphitic fuels poses problems in reactor fueling. At burnups of 40%, the fueling time is about 20 days. It would be difficult, if not impossible, to achieve a plant utilization of 80% if the fueling were done batchwise. Further, in addition to the problem of reactivity control, the variation in distribution of power generation with time would generate severe problems of heat transfer in the core.

These difficulties largely disappear with quasi-continuous fueling. Moreover, if the bi-directional fueling scheme now being actively developed by W. B. Lewis (133) in Canada is adopted, the reactors not only operate in a steady state, but distribution of average concentrations of fuel and fission products are approximately uniform, and the distributions of flux and power generation are symmetrical.

As pointed out by Perry and Preskitt (43), it does not appear desirable to have thorium in the core. When thorium is mixed with the fuel, neutron losses to protactinium are excessive. If thorium were passed through the core separately from the fuel in a time short compared to the mean lifetime of protactinium (40 days), and then held for a longer period, the average exposure of the protactinium might be tolerable. To limit the processing costs, a fertile element would need to be cycled through the reactor many times to build up the concentration of U-233 before being sent to the processing plant. Thus a large inventory of fissile material would accumulate outside the reactor, tending to reduce the fuel yield of the system. Furthermore, the problem of cooling the fertile elements while in the core could assume serious proportions. Finally, the charge-discharge problem would be enhanced by an order of magnitude because of the rapidity of operations.

These difficulties can be avoided by segregating thorium in the blanket. If this is done, somewhat more than half of all neutrons must leak into the blanket if breeding gains in excess of unity are to be realized. This requirement dictates that at least one dimension of the reactor shall be small. A little reflection shows that spherical reactors would not satisfy the heat removal requirements, and that pancake reactors would present difficult problems in reactor vessel and blanket design. Cylindrical reactors having diameters of four to five feet and lengths of 20 feet are more attractive in this regard and permit higher pressure to be used. This geometry was chosen for the present study.

Gas-cooled reactors of several types are under development both in the USA and abroad. For the most part, these utilize natural or slightly enriched uranium, and are designed to produce or recycle plutonium. The use of thorium in some has been contemplated. The Pebble Bed (46, 47) and Turret (48) reactors are similar to the GGBR in objectives and philosophy, except that the breeding goals of these are less ambitious.

Sanderson and Porter Engineers have performed a detailed study of a breeder reactor (46), which "is a two-region thermal breeder, operating on the uranium-thorium cycle, in which all core structural materials are graphite. The reactor working fluid, or coolant, is helium . . . Fuel is in the form of unclad spherical elements of graphite, containing fissile and fertile material. Blanket elements are graphite spheres containing fertile material only . . . The capital costs . . . are estimated at . . . \$254.20 per kilowatt. Production costs . . . are between the limits of 7.46 and 8.07 mills per kilowatt hour, based on a conversion ratio of 0.863." In a subsequent study (47), "Fuel is in the form of ceramic coated uranium-graphite fuel elements, spherical in shape. Indications are . . . the (coolant) system activity will be such that direct maintenance can be performed under controlled conditions. The preferred fuel cycle is thorium- U^{233} . Designing for continuous loading, which can be accomplished in a practical manner with spherical fuel elements, results in a reactor system that will breed sufficient fuel to meet its own needs. By subordinating economy, the system can be designed with a doubling time between 15 and 20 years. Fuel cycle costs of a "hold-on" breeder of 125 Mwe output, are competitive with the lowest fissile fuel costs in this country today."

A serious disadvantage of the Pebble Bed concept is that a void fraction smaller than 0.4 cannot be obtained, and it is difficult to get that low. Cores having moderator-fuel ratios in the desired range (8000) are large, and the pressure is limited to low values, thus limiting the specific power severely. Further, the efficiency of heat transfer between gases and beds of randomly packed spheres is much less than between gases and surfaces aligned parallel to the direction of flow. For a given unit of heat exchanged per unit volume and unit temperature difference, the work expended on forcing the coolant through a randomly packed bed is greater.

In the design proposed by Sanderson and Porter Engineers, the burnup would not be the same for all fuel-spheres; those passing down the periphery of the reactor would experience a lesser exposure than those passing through near the axis of the core. Also, the fuel concentration would decrease away from the charging end, and the fission product concentration would increase. While these assymetries may be advantageous in respect to operation and performance, they certainly make difficult the nuclear and thermal analysis of the system.

While the Pebble Bed concept does possess many advantages of practical nature, especially in respect to continuous fueling and in refabrication of recycle fuel, these are obtained at the cost of impaired nuclear capability. The objectives and scope of the present study include the evaluation of the ultimate capability of the combination uranium-thorium-graphite-gas, and therefore justifies optimism in respect to advancements in technology beyond the present. Thus, in order to improve somewhat the thermal and nuclear capability of the system, and to simplify the calculations, a concept was selected that employs bi-directional loading of flat-plate fuel elements similar to those described by Cottrell (62), and studied by Letourneau and Grimble (51).

The Turret reactor concept was studied jointly by Los Alamos Scientific Laboratory and Sandia Corporation (48). They propose " . . . a high temperature experimental reactor facility . . . for the study of fission product release from an unclad ceramic fuel, and the problems imposed by

the operation of a contaminated coolant system . . . The rotating turret design of the core allows the reactor to be fueled continuously at full power. The expected burnup of 10-50% means that only one to three elements per day need to be reprocessed. Uranium is recovered by burning the spent elements, leaving uranium oxide, which is dissolved in acid and purified."

The specific power planned for the Turret Reactor appears to be low compared to that required for high breeding performance. It is not clear that a fertile blanket could be incorporated successfully into the concept, and the difficulties of designing and operating a large rotating core at the high temperatures and pressures required seem formidable. It was concluded that the Turret concept, in its present stage of development, does not lend itself to breeding.

Perry, Preskitt, Carlsmith, and others at Oak Ridge National Laboratory have studied gas-cooled breeder reactors of various types (43, 44, 45, 50), and discovered the important characteristics of high performance systems discussed above. Similar conclusions were reached independently by Inoue, et al (52) working in Japan. The concept selected for study in the present work was developed by consultation with Perry and his coworkers and by reference to their publications, to a report on the HGCR-1 by Cottrell, et al (62), and to an ORNL Staff Report on advanced concepts of gas-cooled reactors (64). A scheme for processing spent fuel and refabrication of recycle fuel was developed with the aid and advice of members of the Chemical Technology Division, especially J. Guthrie (49).

5.4.2 GGBR Reactor Concept

The concept selected for study has a horizontal, cylindrical core (4 to 5 ft in diameter and 20 ft long) surrounded by a blanket region 2 ft thick, as indicated in Fig. 5.4.1. The dimensions were chosen to give a neutron leakage to the blanket of about 50%, with small axial leakage. A graphite reflector one-ft-thick surrounds the blanket, and this is enclosed by an iron thermal shield one ft thick to protect the pressure vessel, which has walls of carbon steel 10 in. thick.

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Fig.5.4.1 Gas-Cooled, Graphite-Moderated, Breeder Reactor.

Helium enters through an outer concentric pipe at 525°F, 2200 psi and passes along the length of the reactor between the thermal shield and the pressure vessel, passes back through the blanket and core, from whence it is discharged into the inner concentric pipe at 1500°F. Maximum gas velocity is 370 ft/sec. Approximate calculations show that the graphite fuel temperature will probably not exceed 2000°F. It is thought that temperatures up to 4000°F are tolerable (60).

The machinery for charging and discharging fuel elements is located in the ends of the reactor vessel. No work on the conceptual design of this equipment has been done. It must be capable of handling four fuel elements (weighing 20 to 30 lbs) per hour, and of operating at 600°F for long periods without maintenance. The loading machinery must be designed to remove end-blanket elements, charge or discharge a fuel element, and replace the blanket element. The development of such equipment constitutes a major difficulty of this concept.

~~The core region consists of intersecting planes of permanent structural~~ graphite about one in. thick forming square fuel channels (four to six in. inside dimensions) running parallel to the long axis of the core, and is built of premachined sections after the vessel has been set in place. The sections are inserted through the vessel nozzle openings and locked in place by suitable devices. Reactor-grade graphite in sections as large as 16 in. square by 20 ft long can be fabricated by current techniques (61).

The permanent structure in the blanket is also composed of graphite, and occupies 50-75% of the volume. The shape and arrangement of the channels will depend upon the method of cooling adopted. If the cooling can be accomplished through 20 ft channels filled with ThO₂ pellets by axial flow of helium at available pressure drop (equal to that across core), then square channels will be satisfactory, and the blanket structure can be merely an extension of the core structure. If, on the other hand, the cooling requirements are excessive, as in the case where the fertile material is retained for a long time in the blanket before being processed to build up the concentration of U-233, then provision for radial flow of coolant will be necessary. In that event, channels in the form of sectors concentric with the axis of the reactor, as shown in Fig. 5.4.1, would be preferred.

A graphite reflector one ft thick was incorporated in order to reduce leakage of neutrons. This improves the nuclear performance and protects the pressure vessel. The thermal shield is constructed of layers of carbon steel one in. thick laminated with gas-coolant passages. The steel may be borated in order to reduce both the neutron and gamma dose in the pressure vessel. The pressure vessel is designed to operate at 600°F.

Further data for a typical reactor plant are given in Table 5.4.1.

5.4.3 GGBR Fuel System

The fuel elements (Fig. 5.4.1) are composed of unclad graphite plates (0.1 to 0.25 in. thick) impregnated with UO_2 or UC and assembled in open-ended, rectangular, graphite cases (4 to 6 in. square and 2 ft long with walls $\sim 1/2$ in. thick). To reduce neutron streaming in the gas passages, the fuel plates can be set at a slight angle with respect to the long axis of the cases. This eliminates line-of-sight passages through the core.

The fuel elements are loaded, periodically and at full power, into the ends of the reactors, and travel in opposite directions in adjacent channels. In order that the exposure ("burnup") shall be the same for all fuel elements, the residence time in a channel is made inversely proportional to the average fission rate in that channel. As shown in Appendix G, this mode of operation results in nearly uniform axial distribution of the concentrations of fuel isotopes and fission products, when these are averaged over a cross section of the core. At a burnup of 40%, the concentration of U-233 at the midplane is only 3% less than the average concentration at the ends, and the concentration of fission products is 11.2% higher.

This mode of fueling has many advantages. Foremost, of course, is the saving in power capacity that would otherwise be lost in downtime for batch refueling. Second, the distribution of neutron flux and power density are both symmetrical and more uniform than if the fueling were done from one end alone. This improves both the thermal and nuclear performance,

Table 5.4.1. GGBR Typical Reactor Plant Data

General

Station net electrical power, Mwe	1000
Station pumping power, Mwe	69.0
Station plant auxiliaries, Mwe	38.6
Station net efficiency, %	36.1
Station thermal power, Mwt	2770
Reactor thermal power, Mwt	692
Number of reactors per station	4
Core geometry	Cylinder
Core diameter and length, ft	4.6 x 20
Core volume, ft ³	334
Core moderator material and density, g/cc	Graphite; 1.8
Per cent of core occupied by moderator (including fuel plates)	75
Core vessel	None

Blanket thickness, ft	2
Blanket volume, ft ³	1060
Blanket moderator material and density, g/cc	Graphite; 1.8
Per cent of blanket occupied by moderator	50 - 75
Reflector thickness, ft	1
Reflector material and mean density, g/cc	Graphite; 1.8

Thermal shield thickness, ft	1
Thermal shield materials and mean densities, g/cc	Carbon steel; 7.85
System operating pressure, psi	2200
Pressure vessel inside diameter, ft	12.6
Pressure vessel thickness, in.	11
Pressure vessel material	Carbon steel
Pressure vessel design temperature, °F	600

Nuclear

Thermal flux, neutron-cm/cm ³ sec	0.8×10^{15}
Per cent thermal fissions	80
Per cent fissions in fertile stream	10

Table 5.4.1. (Continued)

Fuel Stream

Character	Solid
Fuel form	UO ₂ or UC
Carrier	Graphite
Carrier geometry	Fuel plates
Per cent of core volumes occupied by fuel-carrier	15.9
Carrier mean density, lb/ft ³	112
Fissile atom stream-concentration, atoms/cc	0.5×10^{20}
Plate dimensions, in.	3.4 x 24 x 0.185
Maximum temperature, °F	1900
Maximum temperature rise in fuel-plates, °F	175
Maximum heat flux, Btu/hr ft ²	0.7×10^6
Average heat flux, Btu/hr ft ²	0.3×10^6
Young's modulus, psi	1.6×10^6
Poisson's ratio	0.25
Maximum thermal stress, psi	600
Maximum allowable thermal stress, psi	1000

Core Cooling

Coolant	Helium
Per cent of core volume occupied by coolant	25
Mean density, lb/ft ³	0.556
Mean heat capacity, Btu/lb °F	1.248
Maximum temperature, °F	1500
Minimum temperature, °F	525
Flow rate, ft ³ /sec	750 - 860
Per cent of total heat output removed by core coolant	80 - 90
Velocity in core, ft/sec	300 - 400
Maximum pressure, psig	2200
Pressure drop across core, psi	75 - 90
Circuit pressure drop, psi	100 - 125

Table 5.4.1. (Continued)

Fertile Stream

Character	Solid
Thorium form	Pellets
Pellet dimensions, in.	1/8
Pellet density, lb/ft ³	600
Per cent of blanket volume occupied by fertile stream ^a	15

Blanket Cooling

Coolant	Helium
Per cent of blanket volume occupied by coolant ^b	10 - 35
Mean density, lb/ft ³	0.556
Mean heat capacity, Btu/lb °F	1.248
Maximum temperature, °F	1500
Minimum temperature, °F	525
Flow rate, ft ³ /sec	100 - 200
Per cent of total heat output removed by blanket coolant	10 - 20
Maximum pressure, psig	2200
Pressure drop across blanket, psi	75 - 90
Circuit pressure drop, psi	100 - 125

Steam Plant

Pressure, psig	1450
Temperature, °F	1050
Condenser pressure, in. Hg abs.	1.5
Thermodynamic efficiency, %	40

^aIncluding 40% voids in pellet beds.

^bNot including voids in pellet beds.

for peaking of the flux at one end of the core would have several deleterious effects, among these being increased losses to Pa, increased leakage, and difficult cooling problems.

The proportions in which the available graphite is distributed among the fuel plates, the cases, and the permanent structure may be adjusted to meet the thermal and mechanical requirements (Sec. 6.4.2). The maximum heat flux is about $3/4$ million Btu per hour per square ft, which is not excessive. Maximum thermal stress in the fuel plates (600 psi) appears to be tolerable.

Because of accidental mal-distribution of fuel, coolant, etc., the maximum temperature may get into ranges where UO_2 would react with graphite to form UC_2 and release CO into the coolant. This would metastasize in the colder portions of the circuit, forming CO_2 and depositing carbon. The CO_2 , upon returning to the core, would oxidize more graphite to CO, etc. This transport of carbon from the core to the heat exchanger can be avoided by firing the fuel plates during manufacture to a temperature sufficiently high that UC_2 would be formed. This, however, is unstable toward oxygen, especially in the presence of moisture, and the fired elements would have to be handled in a dry, inert atmosphere. The reactor coolant would have to be continuously purified to remove traces of carbon-reactive elements resulting from accidental contamination.

The problem of radiation damage appears to be more severe than the thermal problems in the fuel. At 40% fuel burnup, about 0.02% of all atoms in the matrix are fissioned. The extent of dimensional changes and deterioration of physical properties (especially thermal conductivity and Young's modulus) at the proposed operating temperature are not known. Experiments conducted by National Carbon Company (77) on a graphite-matrix fuel at 1600°F and exposure of 5×10^{19} fissions/cc of graphite showed negligible damage. The exposure in the GGBR fuel at 40% burnup will amount to only 2×10^{19} fissions/cc. The Turret experiment (48) should throw considerable light on this matter.

A flow diagram of the processing scheme for the GGBR is shown in Fig. 5.4.2. It is proposed to process the spent fuel in the second stage of a Thorex plant (Sec. 5.1.4). Fuel elements discharged from the reactor are held for 90 days for decay of fission product activity, since the organic solvent used in the Thorex plant breaks down under radiation, forming materials which collect at the interface between organic and aqueous phases. This interfacial material interferes with the operation of the extraction plant. Also, it contains uranium, which is lost when the interfacial "gunk" is discharged to waste.

The fuel plates are first separated from the re-usable graphite cases, and treated with fuming nitric acid. This disintegrates the plates and leaches the uranium and soluble fission products. The off-gas is treated to recover radioactive isotopes, and the residue is discharged to the atmosphere. The amount of CO_2 formed is expected to be small.

Due to ingrowth of U-232, the level of activity in the recovered uranium isotopes will be too high for direct refabrication of fuel elements. ~~It is~~ proposed to insert fresh, barren, graphite plates into the re-usable graphite cases and to soak the assemblies in uranyl nitrate solution from the Thorex plant. To prevent impregnation of the cases, they could be coated with wax. The assemblies are fired to 800°C to decompose the uranium nitrate, and volatilize the wax.

A supply of fuel elements sufficient to fuel the reactor for 30 days is reserved.

In the estimation of neutron losses to fission products, no allowance was made for the burnout of isotopes other than xenon and the two samarium isotopes of importance. The effective cross sections assigned to the group of "other fission products" is thus somewhat too high, compensating in part for the use of a mean concentration some 10% too low, as discussed above. Also, no allowance was made for possible escape of fission products (including xenon and iodine) from the fuel into the gas stream and removal from the core. This severely penalizes the GGBR, and positive evidence of removal of an appreciable fraction of the xenon (or iodine) would increase the estimate of its nuclear performance substantially.

FIG.5.4.2 Fuel Processing for Graphitic-Fuel Gas-Cooled Reactors.

Fissile atom loss during reprocessing was estimated at 0.1% per pass through the Thorex plant and 0.1% per pass through the fuel-plate dissolutions process. Every atom of U-233 coming from the fertile stream makes an initial pass through the Thorex plant. On the average at 40% burnup, it makes 1-1/2 passes through the fuel-plate disintegration process and second stage of Thorex processing. Thus, the processing losses are about 0.004 atoms per neutron absorbed in a fissile atom.

5.4.4 GGBR Fertile System

Thorium oxide was selected as the fertile material. It would be desirable to supply the blanket with fertile elements similar in size and shape to the fuel elements so that the same machinery could be used handling them. However, a study of the cost of remotely fabricating plates of ThO_2 showed this to be not economically feasible. In addition, it is necessary, in order to control losses due to absorption of neutrons in Pa-233, to shift the fertile material about in the blanket at least once a day (Sec. 5.1.4). This requirement conceivably could be met by employing ThO_2 in the form of pellets about 1/8 in. in diameter and transporting them into and out of the reactor by gravity flow combined with fluid transport in a stream of helium. The blanket design would probably have to provide for radial flow of cooling gas through the pellet beds in order to avoid exorbitant pressure drops. It is by no means clear that all these provisions are compatible among themselves and with the provision for fuel-element handling.

In order to minimize loss of neutrons to Pa-233, it is desirable to shift irradiated thorium oxide from high- to low-flux zones rapidly. On the other hand, it is desirable to maintain the identity of successive batches of fertile material in order to reduce (approximately by 1/2) the inventory of U-233 in the blanket and the time mean concentration of fission products. Accordingly, it is proposed that the blanket consist of three concentric zones loaded with thorium oxide pellets and separated by graphite structure. Each zone is divided into 33 sectors. Batches of pellets are loaded to the reactor periodically, with one-third of each batch added to

a sector in each zone. The pellets in the innermost sector are shifted every 10 days to the outermost sector, the contents of which are moved to the intermediate zone, and so on. It was estimated that this mode of operation would result in losses to protactinium not more than 5% higher than that computed on the assumption that the blanket is continuously mixed to obtain a uniform concentration of protactinium.

Pellets of ThO_2 removed from the blanket are held 10 days after removal from the reactor to allow activity of fission products to subside. They are then treated in an on-site Thorex plant to remove fission products and recover U-233 (Fig. 5.4.2). It is postulated that Pa-233 (amounting to about 10% of the product in a typical case) will appear with the fission products in the waste. This will be held for 200 days and reprocessed. The protactinium lost will thus amount to less than 0.1% of the bred materials.

Thorium nitrate from the Thorex plant is converted to ThO_2 , pelleted, and recycled to the reactor. These operations must be performed remotely because of the activity of daughters of Th-228.

Neutron losses to fission products formed in the fertile stream were computed for xenon, two samarium isotopes, and a composite group of "other fission products" having a thermal cross section of 50 barns per fission and a resonance integral of 161 barns (24).

Because of the semi-openness of the arrangement at the ends of the core to allow handling of fuel and flow of coolant, somewhat larger additional end losses were taken in the GGBR than in the previous concepts. A loss of 0.004 on the breeding gain was assumed.

5.4.5 GGBR Power System

The power system is similar to that proposed for the HGCR-1 (62). The heat generated in the reactor is removed by circulating helium through one or more external circuits, each consisting of a steam generator, an axial blower, the necessary piping, valves, etc. To alleviate the shaft-seal problem in the blower unit, the entire blower and drive units are enclosed within the primary system volume. The drive unit, separated from

Fig. 5.4.3. Power Generator System for Gas-Cooled, Graphite Moderated Breeder Reactor.

the coolant flow by a thermal barrier, is cooled by an auxiliary system. A continuous purge of helium into the reactor through the container housing the blower unit may be necessary to reduce the problems inherent in operating the motor in a contaminated atmosphere.

Helium flow control will be accomplished by means of bypass and control valves. The bypassed coolant is diverted to the steam generators to remove the heat of compression. In this manner coolant flow within the reactor can be varied without changing the blower operating conditions. A helium purification system operating on a bypass stream is used continuously to remove undesirable coolant contaminants.

Two helium storage systems are required: one for contaminated helium and one for fresh helium. Compressors, some suitable for remote operation and maintenance, are required for the transfer of helium and pressurization of the various coolant systems.

As in the HGCR-1 concept, simple, single-pressure, non-reheat, once-through boilers are used to produce steam at 1050°F and 1450 psia. The steam generators consist of economizer, boiler, and superheater sections in the central region of a vertical unit. The hot coolant enters the top of the unit through the central part of a concentric inlet-outlet arrangement and gives up part of its heat while passing downward through the unit. The cooled gas leaves the boiler and passes through the blower unit and re-enters an annulus surrounding the heat removal section of the boiler. In passing up through the annulus, the coolant serves to control the temperatures of the pressure-containing portion of the boiler.

5.4.6 Status of GGBR Development

The Turret Experiment (48) constitutes a reactor experiment on several of the features important in the GGBR concept presented here. The Turret fueling procedure, involving a rotating core, is not readily adaptable, but the proposed study of fission product release and problems imposed by the operation of a contaminated coolant system are central to establishing the feasibility of the GGBR.

Current designs of gas-cooled reactors employ pressures up to about 300 psi, and the preferred coolant is CO_2 on account of its low cost and high volumetric heat capacity (relative to other gases). The problems of designing equipment to operate with helium at 2000 psi are probably not insurmountable, but will require considerable effort. Leak-tightness is especially important, as helium is expensive. One advantage of operating with the coolant at this high pressure is that if a leak develops in the boiler or superheater the steam, at 1450 psi, will not leak into the helium system.

Considerable effort on the development of graphite dispersions of uranium compounds is presently underway, and should shortly bear fruit. The outlook in regard to radiation damage at levels anticipated in the GGBR is good (77), and rupture by thermal stress does not appear likely (60), although that stress resulting from radiation or thermal gradient induced warping of fuel plates against the restraint of the fuel element case may be a problem. The warping might result from action of radiation and thermal gradients or of non-uniformity of properties in the fuel plates (e.g., coefficient of expansion).

Work is in progress on the fueling of solid fuel reactors at power, and while for the most part this work is concerned with charging fuel elements having lengths equal to that of the core, some of the developing technology should be applicable to the GGBR. The study by Foster et al (106) of the bi-directional loading at power proposed for the CANDU and NPD reactors appears applicable to the GGBR, although the performance requirements are considerably less, and differences introduced by use of cooling with gas rather than water would have to be considered.

The status of development of methods of handling the fertile stream in the GGBR is roughly the same as that in the AHBR. It is proposed to use pellets of ThO_2 in both as fertile material, and cool with helium in the GGBR and with D_2O in the AHBR. All the coolants will be contaminated with fission products. The problems of loading and discharging the blanket are similar, as is also the problem of shifting the fertile material about in the blanket in order to reduce loss of neutrons to protactinium.

The proposed method of preparing the spent fuel in the GGBR for treatment in Thorex has so far not progressed beyond the laboratory stage (82, 83). The manufacture of dispersions of UO_2 or UC in graphite plates by impregnation with uranyl nitrate followed by firing to drive off the oxides of nitrogen has been studied in the laboratory by Sylcor (47), but the problems of performing these operations remotely, and remotely assembling fuel plates into fuel cases, have not been studied. The major developments required for the fruition of the GGBR concept proposed here appear to be:

1. Development of fuel-element handling machinery that can be located inside the ends of a 13 ft diameter pressure vessel at temperatures of 500°F and pressures of 2000 psi.
2. Development of methods of preparing spent fuel for treatment in Thorex, and methods of remotely reconstituting graphite-matrix fuel plates in graphite fuel-element cases.
3. Development of pressure vessels, blowers, heat exchangers, etc., suitable for use with helium as coolant at 2000 psi and 1500°F with a temperature range of 1000°F .
4. Development of methods of batchwise charging, shifting, and discharging pellets of thorium oxide in the blanket that will not interfere with the cooling or loading of the core.

5.5 Deuterium-Moderated Gas-Cooled Breeder Reactor (DGBR)

5.5.1 Introduction

Heavy water, in addition to its obvious utility in homogeneous reactors, may also be used as a moderator and/or coolant in heterogeneous reactors. While it is possible that unclad ceramic or graphitic fuels may be developed which can be cooled by water in direct contact, these are presently in an early stage of development. Further, because of the limitation on heat transfer to water imposed by the necessity of avoiding film boiling, and the limitation on thermodynamic efficiency imposed by the high vapor pressure of water, it turns out that water is not markedly superior to high-pressure gas as a coolant in heterogeneous reactors, especially if a clad is required with water but not with gas. Therefore, it was decided to employ in the DGBR an unclad, graphitic fuel and to cool it with helium. The heavy water functions only as a moderator.

The only structural material presently available for containing heavy water and suitable for use in a high performance breeder is ~~zirconium and its alloys~~. While the parasitic cross section of zirconium is relatively low (~ 0.2 barns), it is appreciable, and it must be used sparingly if a good nuclear economy is desired. The use of pressure tubes was therefore rejected; instead, heavy water is contained in a Zircaloy calandria provided with thin-walled tubes and a pressure vessel is employed.

The moderating power of heavy water is appreciably better at lower temperatures where the density is higher. Although the heat from the moderator may profitably be used for feed-water heating, the use of insulation between the fuel and the tube walls is indicated to keep the amount of heat that must be removed from the moderator within bounds.

As with all high performance gas-cooled breeder reactors, the conclusion is soon reached that the pressure must be as high as possible. This leads to the choice of a cylindrical pressure vessel not exceeding 12 or 13 ft in diameter which will permit pressures up to 2000 psi.

In order to obtain optimum moderator-to-fuel atom ratios, it is necessary to confine the fuel and coolant streams in a relatively small fraction of the core volume. As a result, the fuel concentration in the graphite carrier is very much higher than it is in the GGBR (Sec. 5.4), the heat transfer surface is much smaller, and the heat fluxes are higher. Problems in radiation damage, thermal stress, etc. are therefore much more severe.

The general characteristics of heavy-water moderated gas-cooled breeder reactors have been discussed by Perry (44, p. 13) and others (57, p. 19). Roy and Valerino (72) have described preliminary designs for a CO₂-cooled, pressure tube, heavy-water moderated power reactor scheduled for completion by 1965 by the East Central and Florida West Coast Nuclear Groups (71). The reactor employs slightly enriched uranium oxide fuel elements clad with beryllium and finned to improve heat transfer. The pressure tubes are constructed of Zircaloy and are insulated from the hot gas (~ 1000°F maximum) by ceramic liners containing stagnant CO₂. Conversion ratios up to 0.83 were predicted. Schludi and Plant (68) performed a comparative study of D₂O and CO₂ as coolants, and concluded that the gas-cooled system was superior, mainly because of higher thermodynamic efficiency. Holmes, et al (70), reached the same conclusions independently.

The use of wet steam as a coolant for heavy-water moderated reactors has been studied by Casagrande and Cravarolo (67) and by the Nuclear Engineering Branch, AECL, Chalk River, Ontario (69); the latter concluded that it is not attractive in natural uranium converter reactors.

Related feasibility studies are those of Liljeblad and Madsen (73), Klik and Markvart (75), and Horowitz and Yvon (74).

The concept described in following sections was developed to resolve, as far as possible, the conflicting nuclear, mechanical, and thermal requirements for high performance. A high degree of optimism was employed and many difficult problems of construction and operation that do not bear directly on the nuclear performance or the fuel cost were ignored. The restrictions on core designs are eased a bit if thorium is allowed to be present in the core (though separate from the fuel); the presence of thorium provides

another degree of freedom in achieving an optimum relation between moderation and parasitic captures in heavy water, Zircaloy, and fuel.

5.5.2 DGBR Reactor Concept

The reactor selected for study was a horizontal, cylindrical core region about 7 ft in diameter and 20 ft long (Fig. 5.5.1). This is surrounded by a blanket region about 1 ft thick. The moderator (D_2O containing 0.25% H_2O) is contained in a Zircaloy calandria about 10 ft in diameter and 25 ft long. The core region is penetrated by 36 Zircaloy-2 fuel- and coolant-channels 5.75 in. I.D. with walls 0.125 in. thick arranged in a triangular lattice having an average pitch of 14 in. Graphite sleeves 5.25 in. O.D. with walls $3/16$ in. thick pass through these Zircaloy tubes; the spaces between the sleeves and the metal tubes are filled with a stagnant helium and low density cellular Zirconium oxide or similar material having low parasitic cross sections.

The fuel elements in the channels are cooled by helium which enters at $525^{\circ}F$ and 2000 psi, flows down the annular space between the thermal shield and the pressure vessel, passes back through the fuel channels, and is discharged at $1500^{\circ}F$.

The core region is also penetrated by 36 double-walled Zircaloy tubes containing $1/8$ in. pellets of ThO_2 in the annular space between the porous walls, which permit D_2O coolant to flow radially through the pellets region. The inner tube is $3/8$ in. I.D. with walls $1/8$ in. thick; the outer tube is 2.7 in. O.D. with walls $1/8$ in. thick. Both tubes are pierced by holes $1/16$ in. in diameter which reduce the amount of metal in the tubes by 50%.

The blanket is similar to that of the AHBR described in Sec. 5.1. Pellets of ThO_2 are contained in three concentric annular zones divided circumferentially into 30 "baskets".

The Zircaloy calandria is enclosed in a laminated, gas-cooled, iron thermal shield perhaps 3 in. thick to protect the pressure vessel, which is constructed of carbon steel, and is about 11 ft in diameter and 40 ft long, with walls 9 in. thick.

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D₂O-Moderated, Gas-Cooled Breeder Reactor.

Fig. 5.5.1. Deuterium-Moderated Gas-Cooled Breeder Reactor.

As in the GGBR, machinery for handling the fuel elements is located at the ends of the reactor inside the pressure vessel (Sec. 5.4.2). The general problem is the same.

Additional reactor plant data are given in Table 5.1.1.

5.5.3 DGBR Fuel System

The fuel elements are constructed by nesting several hollow cylinders of graphite about 2 ft long and having walls 1/8 in. thick, as shown in Fig. 5.5.1. The fuel cylinders are impregnated with UO_2 or UC by the process described in Sec. 5.4.3. They are supported at the ends by means of graphite spiders and are mounted in a cylindrical, graphite case which just fits inside a graphite sleeve that passes through the Zircaloy calandria. The diameters of the nested fuel cylinders are chosen so that the temperature rise of the coolant shall be the same along every path through the element.

Fuel elements are loaded and discharged quasi-continuously at both ends of the reactor with counter-current motion in alternate channels. In order to achieve the same exposure (burnup) in all channels, the residence time in a channel is made inversely proportional to the average fission rate in that channel.

The fuel elements are handled, processed, and refabricated by the methods proposed for the GGBR, as described in Sec. 5.4.3. The rate of consumption of graphite is much less than with the GGBR, however. Other aspects (fission product poisoning, processing losses, etc.) were assumed to be the same. No allowance was made for possible escape of xenon, etc. from the fuel elements into the gas stream.

Because of the heavy loading of fuel in the graphite (0.24 gms U/gm graphite), the radiation damage to graphite will be more severe than in the GGBR. The exposure in the DGBR at 40% burnup of fuel atoms will amount to 3.7×10^{20} fissions/cc, which is considerably larger than the 5×10^{19} fissions per cc in a test sample reported by the National Carbon Company (19).

Table 5.5.1. DGBR Typical Reactor Plant Data

General

Station net electrical power, Mwe	1000
Station pumping power, Mwe	65
Station plant, auxiliaries, Mwe	35
Station net efficiency, %	32.2
Station thermal power, Mwt ^a	3100
Reactor thermal power, Mwt	775
Number of reactors per station	4
Core geometry	Cylindrical
Core diameter and length, ft	7.3 x 20
Core volume, ft ³	830
Core moderator material and density, g/cc	D ₂ O; 1.02
Per cent of core occupied by moderator (excluding graphite)	80.8
Fuel-tube thickness, in.	0.125
Fuel-tube material and weight, lb	Zr; 6000
Per cent of core occupied by structure ^b	6.2
Blanket thickness, ft	1.0
Blanket volume, ft ³	655
Blanket moderator material and density, g/cc	D ₂ O; 1.02
Per cent of blanket occupied by moderator	83.9
Reflector thickness, ft	0.18
Reflector material and mean density, g/cc	D ₂ O; 1.02
Thermal shield thickness, in.	3
Thermal shield materials and mean densities, g/cc	Iron; 7.85
System operating pressure, psi	2000
Pressure vessel inside diameter, ft	11
Pressure vessel thickness, in.	9
Pressure vessel material and clad	Carbon steel, S.S.
Pressure vessel design temperature, °F	600

^aIncludes waste heat in blanket coolant.

^bIncludes Zircaloy in tubes, ZrO insulation, graphite sleeve.

Table 5.5.1. (Continued)

Nuclear

Thermal flux, neutron-cm/cm ³ sec	0.75 x 10 ¹⁴
Per cent thermal fissions	82
Per cent fissions in fertile stream	10 - 20

Fuel Stream

Character	Solid
Fuel form	UO ₂ or UC
Carrier	Graphite
Carrier geometry	Concentric cylinders
Per cent of core volume occupied by fuel carrier	1.9
Carrier mean density, lb/ft ³	112
Uranium mean-concentration in fuel stream, atoms/cc	0.8 x 10 ²¹
Element wall thickness, in.	0.125
Element dimensions, in. (length x diameter)	24 x 5
Maximum temperature, °F	2600
Maximum temperature rise in fuel, °F	826
Maximum heat flux, Btu/hr ft ²	1.6 x 10 ⁶
Average heat flux, Btu/hr ft ²	0.7 x 10 ⁶
Young's modulus, psi	2.9 x 10 ⁻⁶
Poisson's ratio	0.25
Maximum thermal stress, psi	1100
Maximum allowable thermal stress, psi	1100
Mean moderator temperature, °F	300

Core Cooling

Coolant	Helium
Per cent of core volume occupied by coolant	9.2
Mean density, lb/ft ³	0.505
Mean heat capacity, Btu/lb °F	1.248
Maximum temperature, °F	1500
Minimum temperature, °F	525
Flow rate, ft ³ /sec	834
Per cent of total heat output removed by core coolant	80 - 90
Velocity in core, ft/sec	441
Maximum pressure, psig	2000

Table 5.5.1. (Continued)

Core Cooling (Continued)

Pressure drop across core, psig	69
Circuit pressure drop, psig	92

Fertile Stream

Character	Solid
Thorium form	ThO ₂ pellets
Pellet dimensions, in.	1/8
Pellet mean density, lb/ft ³	600
Per cent of blanket volume occupied by fertile stream ^a	16.1
Per cent of core volume occupied by fertile stream ^a	2.4
Carrier form	Pellets

Blanket Cooling

Coolant	D ₂ O
Per cent of blanket volume occupied by coolant ^b	84
Mean density, lb/ft ³	63.6
Mean heat capacity, Btu/lb °F	1.02
Maximum temperature, °F	305
Minimum temperature, °F	295
Flow rate, ft ³ /sec	162
Per cent of total heat output removed by blanket coolant	21
Maximum pressure, psig	2000

Structure

Material, and weight, lb	Zr, 7250
--------------------------	----------

Steam Plant

Pressure, psi	1450
Temperature, °F	1050
Condenser pressure, in. Hg abs.	1.5
Thermodynamic efficiency, %	40

^aDoes not include 40% voids in pellet beds

^bIncludes D₂O in voids in pellet beds

The limiting design condition of the fuel element is the thermal stress in the graphite. At 2600°F the thermal stress in the DGBR fuel plates is 1100 psi, which is a factor of 2 below the tensile strength at this temperature. Since the DGBR is gas-cooled, and there is no cladding on the fuel element, the limiting condition on the heat flux is that which will maintain the thermal stress within design limits. For this reactor, the maximum heat flux is $1.6 \times 10^6 \text{ Btu/hr-ft}^2$.

5.5.4 DGBR Fertile System

The distribution of pellets of ThO_2 in the core and blanket of the reactor is described above in Sec. 5.5.2. They are loaded into, shifted about, and removed from the reactor by hydraulic transport in streams of D_2O as described in reference 9. Three baskets, one in each zone, or two baskets and a pellet tube in the core, constitute a set containing a discrete batch of pellets. A new batch is loaded every 5 to 10 days, giving cycle times of 150 to 300 days. The pellets in a given batch are shifted every 5 - 10 days from one basket to another in order to reduce the average exposure of the protactinium. As in the AHBR, the pellets are cooled by radial flow of D_2O , which enters the reactor at 295°F , and is discharged at 305°F . About 260 Mwt of the heat removed by the fertile stream coolant can be used for feed-water heating; the rest must be wasted. In this concept then, the power removed from the core remains constant, and the overall efficiency decreases as the loading of fissile isotopes in the blanket discharge is increased.

It is proposed to process the ThO_2 pellets in an on-site Thorex plant as described in Sec. 5.1.4. The pellets are held for 10 days after removal from the blanket system to allow activity of fission products to decay to an acceptable level, before processing. The fission products, containing undecayed Pa-233, are held for 200 days and reprocessed to recover U-233.

The assumptions regarding fission product poisoning, processing losses, etc., were the same as for the AHBR.

5.5.5 DGBR Power System

The power system proposed for the GGBR (Sec. 5.4.5) is suitable for the DGBR, and was adopted without change.

In the core, improvement in heat removal can be effected by distributing the fuel-coolant channels in an approximation of the Goertzel loading (90), whereby the channels are clustered more closely near the axis of the core and are scattered more sparsely near the periphery. Fast neutrons tend to migrate from the under-moderated central region and to be slowed down in the over-moderated peripheral regions. As a result, the distribution of the thermal flux becomes nearly uniform, and the power generated in every channel is the same. Every channel will operate at maximum capacity, the temperature rise in the coolant will be uniform, and there is no need to orifice peripheral channels.

5.5.6 Status of DGBR Development

The status of development of fuel, fuel handling, and fuel processing for the DGBR is roughly the same as for the GGBR (Sec. 5.4.6). The performance required of the fuel is greater, with heat fluxes approximately twice as great, and the radiation exposure (fissions/cc) 20 times as great. The fuel element handling problem is more difficult in that the elements are considerably more fragile (to eliminate excess graphite) and the elements must be loaded into and discharged from fuel channels intermingled with other channels charged with pellets of thorium oxide. In addition, the fuel tubes must be provided with insulation between the hot cooling gas and the relatively cool heavy-water moderator, and the whole must be supported in calandria tubes containing a minimum of Zircaloy. Although an analysis of hazards lay outside the scope of the present study, it is clear that they are greater with DGBR than with the GGBR. The reactor currently under construction by the American Electric Power Service Corporation (71) will serve to a certain extent as a demonstration of these developments.

The major development problems in the DGBR thus include all those listed for the GGBR, and in addition, the perhaps more difficult problem of designing for nuclear and thermal compatibility among the fuel, coolant, structure and moderator.

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6. FUEL CYCLE OPTIMIZATION

6.1 Aqueous Homogeneous Breeder Reactor (AHBR)

In this section, the results of several parametric studies directed toward determining the maximum fuel yield obtainable in the various concepts at various fuel cycle costs ranging from 1.5 mills/kwhr. Parameters which did not affect the fuel cycle cost, or to which they were not sensitive, were set at their optimum or limiting values insofar as these could be estimated. For example, the pressure in the gas-cooled reactors was limited by the allowable stress, wall thickness, and diameter of the pressure vessel. The moderator-fuel ratio is similarly determined by nuclear rather than fuel cost considerations.

Other parameters, with respect to which the fuel yield was not sensitive, were set at their fuel-cost-optimum or limiting values. For example, the use charge on D_2O in the DGBR is charged against the fuel cycle cost, but the amount of D_2O in the piping, pumps, and heat exchangers of the moderator-cooling circuit has no bearing on the fuel yield. It was, accordingly, made as small as possible, without much regard to the corresponding increase in capital costs.

The independent parameters which usually turned out to be important in determining both the fuel yield and the fuel cycle cost were the following: (a) fuel stream processing rate, (b) fertile stream processing rate, and (c) thorium inventory. As indicated in Sec. 3, parasitic captures in fission products and protactinium, inventories of U-233, breeding credit, etc., are complex functions of these parameters. The complex relation, however, can be reduced to a rather simple relation between fuel cycle cost and fuel yield, as illustrated in Fig. 3.2. This procedure was used in the optimization of the fuel costs.

6.1.1 AHBR Special Assumptions

Critical diameters were determined by means of a one-dimensional, 34-group calculation (GNU program) in cylindrical geometry with a transverse buckling corresponding to a length of 12 ft. In these calculations, it was not possible to take the end-blankets into consideration. While most of the neutrons leaking in the axial direction are absorbed in the end-blankets, there will be some reflections, and the effect is to render the reactor critical with: (a) a shorter length, (b) a smaller diameter, or (c) a lower concentration of fuel in the core. The effect of all these compensatory changes is to increase the overall leakage to the blankets. Thus, in effect, the neutrons calculated to leak from a bare-ended reactor are captured somewhere in the end or side blankets. Accordingly, the neutron balance resulting from the calculation of the bare-ended reactor was adjusted by allocating the axial leakage among the various possible blanket events (absorption in thorium, Zr, etc.) in the same proportions that neutrons leaking radially into the side blankets were estimated to be distributed by use of the equilibrium equations.

Some of the delayed neutrons will be emitted outside the core in the piping, pumps, heat exchangers, etc. This loss was estimated by the method described by Walker (20) with data taken from a compilation by Leachman (21). The mean transit time in the core and external circuit was based on the assumption of slug-flow.

The capture of neutrons by protactinium and U-233 in the blanket was based on the approximation that the concentration of these is uniform throughout the fertile material. The error is greatest in connection with the estimation of the losses to protactinium, but was found to be less than 1% when the cycle time for shifting the fertile pebbles from one zone to another is of the order of three days (see Appendix D).

In this study, the D_2O , is an important factor in the fuel cycle cost. The station inventory is approximately 200 tonnes, of which about 2/3 is associated with the fertile stream. A use charge of 12.7%/yr and a replacement rate of 3%/yr were assumed.

6.1.2 AHBR Key Variables

Levels of rare-earth fission-product poisoning (Sec. 5.1.4) attainable at convenient and economic rates of processing of fuel solution through the hydroclones are so low that further reduction is not practical. The loss of neutrons to solubles and to xenon is also controlled to very low levels with little effort or cost. The volume of the fuel stream in the external portions of the circuit was minimized as described in Sec. 5.1.3. The rate of processing the uranium in the fuel stream is determined entirely by the composition and weight-rate-of-flow of solids from the hydroclones. Thus, in regard to the fuel system, the only variables that appear to be not determined from previous work are the optimum moderator-fuel ratio and the corresponding critical diameter of the core vessel.

The performance of the system as a whole is related to the fuel concentration. As this is lowered, the fraction of fissions caused by thermal neutrons increases, and the mean value of η increases. The inventory decreases also. These gains are offset by an increase in parasitic captures in hydrogen, deuterium, fission products, corrosion products, and Zr.

The general shape and size of the reactor core was readily established from previous work (Sec. 5.1.2). It was known that, at a diameter of about 4-1/2 ft, the critical concentration of U-233 is about 1.5 grams per liter (13). Concentrations of 1.0, 1.5, and 1.8 grams per liter were selected for study, and the critical diameters (~~corresponding to equilibrium concentrations~~ of higher isotopes, fission products, etc.) were calculated by means of the GNU and ERC programs for IBM-704 (Sec. 3). As discussed in Sec. 6.1.3, the calculations show that the performance, as measured by the fuel yield and the fuel cycle cost, is insensitive to variations in fuel concentration in the range studied. Accordingly, a concentration of 1.5 grams of U-233 per liter was used in subsequent calculations.

This insensitivity confers freedom in selecting lengths and diameters for the cores such that the total heat requirement of the station may be supplied by an integral number of reactors, with each reactor operating at full capacity. Rather large changes in length, to which the power is directly proportional, can be compensated by relatively small changes in concentration with little effect on the performance.

A value of 50 watts/cc was adopted for the maximum allowable power density in the fuel solution adjacent to the core-vessel wall. The rate of attack on the Zircaloy wall, based on HRT experience, is thought to be tolerable at this level for a period of 10 years*(14). In cores 4-1/2 ft in diameter and 12 ft long, the ratio of average power density to maximum power density at the wall is about three, so that average power densities of about 150 kw/liter and heat rates of 800 Mwt per core may be realized. The lengths may be adjusted so that an integral number of reactors will produce 3600 Mwt. Thus, either four reactors ~ 15 ft long, or five reactors ~ 11 ft long would suffice. From the standpoint of capital investment, the preferred number of reactors is four.

In regard to the fertile system, the parameters of importance are the thorium inventory and the processing cycle-time. When the cycle-time is long compared to the mean lifetime of protactinium (40 days), the inventory of protactinium in the blanket is approximately equal to forty times the daily production rate. In reactors having breeding ratios in the range 1.0 to 1.1, the production rate is about equal to the rate of burnup of fuel, and this is directly proportional to the power, since about one megawatt of heat is produced per gram of fuel fissioned. To produce 1000 Mwe with an efficiency of 27.5% requires a heat rate of 3600 Mwt. The inventory of protactinium in the blankets is thus about 150 kg, irrespective of the amount of thorium present. Since the protactinium is mixed with the thorium, the rate of capture of neutrons by protactinium may be decreased by increasing the thorium inventory. But this results either in an increase in the processing rate or an increase in the inventory of U-233 in the blanket, or both. The optimum combination of thorium inventory and blanket-processing cycle-time is that which gives the lowest sum of processing costs, inventory charges on thorium and U-233, and value of fuel lost due to capture of neutrons by protactinium.

A fertile-system processing cycle time of about 200 days was found to result in a product stream containing about 6 grams of U-233 (and undecayed Pa-233) per kg of thorium. By means of preliminary calculations, the optimum thorium inventory for this cycle time was found to be about 150 tonnes. The calculations reported in Table 6.1.1 were based on an inventory of 137.5 tonnes

* Replacement cost was not included in fuel cycle cost, but amounts to less than 0.1 mill/kwhr.

Table 6.1.1.

AHBR Nuclear Characteristics

Station Power: 1000 Mwe Fuel Processing: Continuous, on-site

	Case Number						
	1	2	3	4	5	6	7
	Key Variables ^a						
Moderator-fuel ratio, $N(D)/N(U)$	12,000	15,000	12,000	8,500	10,000	10,000	10,000
Reflector thickness, cm	6	6	6	6	20	20	20
Fertile stream processing cycle-time, days	135	218	212	217	395	495	662
	Nuclear Economy						
Net neutron yield, $\bar{\eta}$	2.2544	2.2556	2.2526	2.2514	2.2511	2.2497	2.2514
Nuclear losses ^b :							
U-233 absorptions	0.9180	0.9106	0.9128	0.9141	0.9077	0.9041	0.9040
U-235 absorptions	0.0820	0.0894	0.0872	0.0859	0.0923	0.0959	0.0960
Th-232 fissions	0.0012	0.0020	0.0020	0.0021	0.0036	0.0044	0.0054
Pa-233 absorptions (x 2)	0.0312	0.0363	0.0387	0.0400	0.0442	0.0446	0.0458
U-236 absorptions	0.0060	0.0075	0.0071	0.0069	0.0087	0.0096	0.0104
Np-237 absorptions	0.0017	0.0013	0.0013	0.0013	0.0013	0.0012	0.0011
Xenon absorptions	0.0063	0.0075	0.0075	0.0075	0.0095	0.0103	0.0115
Samarium absorptions	0.0095	0.0115	0.0095	0.0079	0.0082	0.0080	0.0079
Other fission product absorptions	0.0092	0.0116	0.0097	0.0084	0.0125	0.0144	0.0197
Corrosion products absorptions	0.0022	0.0026	0.0020	0.0017	0.0017	0.0016	0.0014
Core moderator absorptions	0.0336	0.0392	0.0315	0.0262	0.0266	0.0247	0.0223
Fuel carrier absorptions	0.0294	0.0290	0.0287	0.0270	0.0244	0.0229	0.0211
Th carrier absorptions ^c	0.0036	0.0033	0.0033	0.0033	0.0028	0.0026	0.0024
Loss of delayed neutrons	0.0071	0.0106	0.0096	0.0116	0.0157	0.0191	0.0242
Leakage	0.0010	0.0010	0.0010	0.0010	0.0010	0.0010	0.0010
Allowance for additional end-losses	0.0010	0.0010	0.0010	0.0010	0.0010	0.0010	0.0010
Processing losses	0.1114	0.0912	0.0997	0.1052	0.0900	0.0843	0.0762
Net breeding gain			Other Characteristics				
	31,600	39,500	31,600	26,400	30,900	30,900	30,900
$\Sigma_s/N(U)$, barns/atom (U-233+U-235)	93	93	92	91	89	88	85
Per cent thermal fissions	8	14	15	14	26	31	38
Per cent fissions in fertile stream							

^a Fertile stream thorium inventory: 137.5 tonnes.

^b Neutrons or fissile atoms lost per neutron absorbed in fissile atom.

^c Includes Zr in core vessel and blanket, D₂O and H₂O in coolant.

in the blanket, and with various amounts up to 13 tonnes in the processing plant. The processing cycle time was varied over a range extending up to about 700 days. The effect of optimizing the thorium inventory at the longer cycle times was studied by perturbation of the calculations.

The effect of varying the thickness of the outer annulus of D_2O was studied briefly. Optimization of the thickness of the D_2O annuli between thorium bearing zones was not studied, but some degree of improvement might be obtained thereby.

6.1.3 AHBR Results

Key variables for the cases selected, together with neutron economies of the corresponding equilibrium reactors are listed in Table 6.1.1. Inventories and processing rates are given in Table 6.1.2, and a fuel cycle cost breakdown in Table 6.1.3.

The concentration of U-233 in the fuel stream was varied in cases 2-4. Increasing the fuel concentration from 1.2 to 1.8 grams per cc results in a decrease of neutron losses to fission products, corrosion products, and fuel carrier. Loss to protactinium increases slightly and η decreases somewhat. The result is a net increase in breeding gain. The inventory of fissile material increases also but not as rapidly as the breeding gain, and the fuel yield increases slightly from 12.4 to 13.2% per year.

A fuel concentration of 1.5 grams per liter ($N(D)/N(U) \sim 31,000$) was used in subsequent calculations. Cases 1, 3, 5, 6, and 7 form a set in which the fertile-stream processing cycle time progressively increases from 135 to 662 days. As a result, the concentration of U-233 and the fission rate in the blanket increase. Loss of neutrons to fission products and to leakage increases, although it is possible to control the latter somewhat by thickening the reflector as in cases 5, 6, and 7. Captures by protactinium increase, and result in the formation of more U-234. This, when transferred to the fuel stream, generates more U-235, which lowers the mean η of the reactor and in turn generates more U-236.

Table 6.1.2. AHRB Inventories and Processing Rates

	Case Number						
	1	2	3	4	5	6	7
	Key Variables						
Moderator-fuel ratio, $N(D)/N(U)$	12,000	15,000	12,000	8,500	10,000	10,000	10,000
Fertile stream processing cycle time, days	135	218	212	217	395	495	662
	Station Inventories						
Station thorium inventory, tonnes	159	150	151	150	145	143	142
Station D_2O inventory, tonnes	201	179	202	220	230	228	229
Station fissile inventory, kg							
U-233 in fuel stream ^a	190	144	173	201	147	132	118
U-235 in fuel stream ^a	18	16	19	21	20	20	20
U-233 in fuel-stream processing	55	43	50	56	41	36	32
U-235 in fuel-stream processing	5	5	5	6	6	5	5
U-233 in fuel-stream reserve	111	97	98	97	81	74	64
U-235 in fuel-stream reserve	10	11	11	10	11	11	11
U-233 in fertile stream	153	293	291	293	565	704	896
Pa-233 in fertile stream	127	143	144	146	157	159	161
U-233 in fertile-stream processing	197	130	134	132	85	71	56
Pa-233 in fertile-stream processing	53	31	33	32	17	14	10
Total, kg per station	919	913	958	994	1130	1226	1373
	Processing Rates						
Spent fuel processing rate ^b , kg of U/day	0.320	0.261	0.286	0.292	0.221	0.180	0.164
Excess fissile uranium produced, kg/day	0.372	0.305	0.334	0.352	0.302	0.284	0.257
Thorium processing rate, kg/day	1065	631	650	634	350	278	222
Thorium replacement rate, kg/day	24.9	16.2	16.7	16.4	10.6	9.1	7.7
D_2O replacement rate, kg/day	16.5	14.7	16.6	18.1	18.9	18.8	18.8
Fertile stream loading, g(U-233+Pa)/kg Th	4.0	6.0	6.0	6.0	9.0	10.5	12

a. Includes allowances for dump tanks and uranium associated with scale on piping, etc.

b. Includes only U-233 and U-235 in underflow from hydroclones.

Table 6.1.3. AHBR Fuel Costs and Performance Summary

	Case Number						
	1	2	3	4	5	6	7
	Key Variables						
Moderator-fuel ratio, N(D)/N(U)	12,000	15,000	12,000	8,500	10,000	10,000	10,000
Fertile stream processing cycle-time, days	135	218	212	217	395	495	662
	Cost Breakdown ^a , mills/kwhr						
Uranium inventory	0.079	0.078	0.082	0.085	0.097	0.105	0.118
Thorium inventory	0.078	0.074	0.074	0.074	0.071	0.070	0.070
D ₂ O inventory	0.225	0.200	0.226	0.246	0.256	0.255	0.255
Fuel-stream processing	-	-	-	-	-	-	-
Fertile-stream processing	1.015	0.849	0.849	0.849	0.700	0.654	0.609
Thorium replacement	0.037	0.023	0.023	0.023	0.015	0.013	0.011
D ₂ O replacement	0.053	0.047	0.053	0.058	0.061	0.060	0.060
Breeding credit ^b	0.291	0.239	0.262	0.276	0.236	0.222	0.201
	Performance Summary						
Net fuel cycle cost ^b , mills/kwhr	1.20	1.03	1.05	1.06	0.96	0.94	0.92
Fuel yield ^b , % per year	14.9	12.3	12.9	13.1	9.8	8.5	6.8
Fuel specific power, Mwe/kg	1.09	1.10	1.04	1.00	0.89	0.82	0.73
Thorium specific power, Mwe/tonne	6.3	6.7	6.6	6.7	6.9	7.0	7.0
Net breeding gain (rounded)	0.11	0.09	0.10	0.10	0.09	0.08	0.08
Per cent of heat from fertile stream	12	18	25	19	29	33	40

a. See Sec. 3.2 for cost bases. No allowance for waste disposal.

b. Plant factor: 80%. The net fuel cycle costs exclude the cost of replacing the core vessel at 10 year intervals. If included in the fuel cycle cost, this would add less than 0.1 mill/kwhr.

There are minor gains, but they do not compensate for the losses, and the breeding gain declines steadily as the fertile stream cycle time increases. This decline, together with a continuous rise in blanket inventory of fuel isotopes (Table 6.1.2) results in the fuel yield falling from 15 to 7% per year in the range investigated.

Case 4 gave the largest breeding gain (0.112). The two largest parasitic losses were those to protactinium (~ 0.03) and to thorium carrier (~ 0.03), which last is mostly due to captures in the Zircaloy of the core-vessel wall and blanket structural members. The losses to protactinium could be decreased by increasing the thorium inventory, but the inventory of U-233 would also increase and the fuel yield would increase only a small amount. Thus, doubling the thorium inventory would reduce the loss to protactinium by about 0.014 and to leakage by 0.005 units. Eta would increase by 0.004 (compare cases 4 and 6 where the protactinium losses differ by about 0.015), and the loss to U-236 might decrease by 0.001, giving a total saving of 0.025 on the breeding gain. But the blanket inventory of U-233 would double, bringing the total inventory from 919 to 1069 kg, and the fuel yield would increase only from 15.1 to 15.9% per year.

It does not appear to be feasible to reduce the structural metal in the blanket appreciably. The blanket may be overmoderated, and some improvement in fuel yield (through reduced losses to D_2O , H_2O , and Zr) and saving in D_2O inventory charges might result from reducing the thickness of the layers of D_2O separating the fertile zones. However, it seems clear that the fuel yield cannot be increased much above 16% per year by these means.

Fuel cycle costs are listed in Table 6.1.3, together with a summary of performance characteristics. The largest contribution is the capital charge on the Thorex plant, and this increases rapidly as the fuel yield increases (Fig. 6.1.1). Further increase in the fertile-stream processing rate would cause the fuel yield to pass through a maximum because improvement of the breeding gain is limited while the fissile inventory continues to rise indefinitely. The fuel cycle cost associated with a yield of 16% is probably not greater than 1.5 mills per kwhr, as indicated in Fig. 6.1.1.

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The minimum cost computed was 0.92 mill/kwhr (Case 7) at a fuel yield of about 7% per year. Nearly 40% of the total reactor heat must be removed by the blanket coolant, and it does not appear to be feasible to increase the fertile-stream processing cycle time further. The cost probably rises at lower yields anyway, because the breeding gain falls more rapidly than the fertile-stream processing cost.

Possibly as much as 0.05 mill/kwhr might be saved by reducing the D_2O inventory in the blanket, at the cost of increased leakage of neutrons, but it will be difficult to reduce costs much below 0.9 mill/kwhr.

It can be noted from Table 6.1.3 that the low fuel cycle cost of 0.9 mill/kwhr is achieved by reducing the fertile-stream processing rate as much as is possible from the standpoint of blanket heat generation. Such a low processing rate has two deleterious effects on fuel costs. First it allows the fissile uranium to build up in the blanket, increasing inventory costs, and second it reduces the net breeding gain. However, even though the unit cost of processing thorium in a Thorex plant increases with decreasing plant capacity, the over-all effect of lower thorium processing rates is lower fuel cycle costs. ~~The reason for this is that the unit processing costs are inversely proportional to the 0.7 power of the processing rate, whereas the total amount of material handled is proportional to the rate. Therefore, the fuel cycle costs increase only as the 0.3 power of the processing rate.~~

This is illustrated in the following table which compares fuel cycle costs for the maximum and minimum processing rates considered in this study with an estimate prepared for the AEC by Sargent and Lundy (131) of the fuel cycle cost in a 316 Mwe, aqueous-homogeneous reactor plant having a solution core and a continuously-processed slurry blanket. As pointed out in Sec. 5.1.4, the concentration of bred material in the discharge stream from a batch-processed blanket is approximately twice the mean concentration in the blanket. In a continuously-processed blanket, on the other hand, the discharge concentration is the same as the mean concentration. As a result, batch-processed blankets have the advantage that, for a given inventory of bred material in the blanket, and a given loading in the discharge stream (grams or bred material per kg of thorium), the processing rate, in kg per day, with the batch-processed blanket is only half as great as with the

Table 6.1.4 Comparison of Fuel Cycle Costs for
Varying Thorium Processing Rates

	Case 1	Case 7	S & L 1674
Net Plant Capacity, Mwe	1000	1000	316
Station Efficiency	27.5	27.5	27.7
Loading, gm U-233 + Pa/kg Th	4	12	4
Thorium Processing Rate, kg/day	1065	222	300
Unit Processing Rate, kg/MwD	1.065	0.222	0.95
Thorium Processing Costs, \$/kg	23	66	49
Fuel Cycle Costs, mills/kwhr			
Uranium Inventory	0.08	0.12	0.24
Thorium Inventory	0.08	0.07	0.16
D ₂ O Inventory	0.23	0.26	
Processing	1.02	0.61 ^a	1.94 ^b
Thorium Replacement	0.04	0.01	
D ₂ O Replacement	0.05	0.06	
Breeding Credit	- 0.29	- 0.20	- 0.22
Net Fuel Cycle Cost	1.20	0.93	2.12 ^c

- a. Batch-processed pellet-blanket (Sec. 5.1.4), with 38% total fissions in blanket.
- b. Continuously processed slurry blanket (Sec. 5.1.4) with ~ 7% of total fissions in blanket.
- c. Excluding D₂O costs.

continuously processed blanket, and consequently, other factors being the same, the fertile stream processing cost will be only half as great.

The processing cost cited in the Sargent and Lundy study is based on an estimate prepared by the Fluid Fuel Reactors Task Force (TID-8507) of \$14.8 million for a 300 kg/day on-site Thorex plant to service a 333 Mwe (gross) and 316 Mwe (net) two-region thorium breeder reactor. At 14% annual fixed charges plus 15% annual operating and maintenance costs, this estimated capital cost results in a net fuel processing cost of 1.94 mills/kwhr. This plant was later stated by the AEC in TID-9518(3), Book 3, to be capable of handling 500 kg of thorium plus 2 kg of fissionable material per day; however, the processing costs were not adjusted in this latter report to allow for the increased processing capacity.

The lack of definite information based on actual construction and operating cost data for Thorex plants of varying capacities leads to a considerable uncertainty concerning processing costs. Estimates made over the last five years by different investigators vary by almost a factor of two for plants of the same capacity and also exhibit varying trends relative to changes in capacity. ~~In general, these estimates of the costs of processing thorium-fuel mixtures in a single-purpose on-site Thorex plant fall in a band whose upper limit is represented by the equation, \$/kg thorium = 3100 R^{-0.7}, where R = plant capacity in kilograms of thorium per day, and whose lower limit by the equation, \$/kg = 1600 R^{-0.7}.~~ For plants with capacities above 2000 kg/day, a fixed charge of \$4/kg of thorium should be added to the above estimates to be consistent with published data. Processing costs reported in this study are based on the highest estimates of on-site Thorex costs in order to allow for escalation of early estimates and to be conservative. Thus the unit processing costs are essentially the same as estimated by the Fluid Fuel Reactors Task Force without any allowance for possible increases in capacity. The difference in processing costs between Case 1 of this study and the Sargent and Lundy estimate, therefore, results partly from the difference in fertile stream loading and partly from the difference in blanket management, rather than from a difference in degree of conservatism regarding processing costs.

The higher breeding credit shown in Table 6.1.4 for Case 1 does, however, represent a change in technology as it is based on more recent (and more correct) values of η of U-233, lower values for the thermal cross section of Pa-233, and slightly greater optimism with respect to removal of xenon; and, it is believed, represents a better optimization of the many factors which influence fuel cycle costs.

6.2 Molten-Salt Breeder Reactor (MSBR)

6.2.1 MSBR Special Assumptions

The nuclear characteristics of a cylindrical core can be calculated to a satisfactory degree of approximation in spherical geometry (one-dimensional) by means of the 3⁴-group GNU program for the IBM-704 (Sec. 3). An equivalent, spherical core having the same average atomic composition as the heterogeneous cylindrical core of the reactor and an equivalent diameter 1.09 times greater than the actual cylinder was first calculated to determine approximately the thorium and protactinium concentrations (i.e., volume fraction of fertile stream in core) in the critical homogeneous reactor. The heterogeneity was then treated by the use of the Cornpone code (91) for the ORACLE. Cornpone calculations of a 6-region unit cell in the core region generated disadvantage factors for fuel stream, fertile stream, and moderator; these factors were then used in subsequent Cornpone calculations to calculate the critical heterogeneous reactor. Criticality was achieved by varying the amount of thorium (volume fraction of fertile stream) in the reactor core.

Reaction rate coefficients for each element in each region were then computed from the neutron balances. These, together with the other required input information (power, volume of fuel and fertile streams, etc., as described in Appendix H) were used to compute the equilibrium composition and performance of the reactor by means of the ERC-5 program for the IBM-704 (86).

The unit cell selected to derive group-disadvantage factors consisted of a bayonet fuel-tube, fertile stream annulus, and associated moderator, and had the following cylindrical regions (listed from the center outward, with thickness given in in.): fuel - 1.19 (radius); graphite - 0.25; fuel - 0.43; graphite - 0.5; fertile stream - 0.25; graphite - 1.6.

Neglecting the 10 volume % of graphite structure, the blanket of the MSBR is nearly homogeneous except where the fuel tubes pass through it above the core. This upper end-blanket was homogenized with the radial and lower end-blanket; weighting factors were taken proportional to the interfacial areas at the surface of the core.

Delayed neutrons were estimated by the method described by Walker (20).

As a result of discussions with the Y-12 production department of UCNC in Oak Ridge, it was concluded that a supply of elemental lithium having a composition 99.99% Li-7 could be postulated at reasonable cost and in sufficient quantity to support a large number of molten-salt reactors. The Li-6 (0.01%) tends to burn out, and approaches an equilibrium concentration that is a function of the feed rate, the discard rate, and the rate of reaction with neutrons. By means of an auxiliary calculation, using reaction rate coefficients calculated from GNU results, the equilibrium concentration of Li-6 in the fuel stream was found to be about 0.0025%. Thus, about 2/3 of the Li-6 atoms capture neutrons and the rest are removed in the discard of fuel salt.

The formation of Li-6 by the n, α reaction in beryllium was neglected. Hafele (40) has shown that the n, α and the $n, 2n$ reactions in beryllium just about cancel each other in respect to neutron losses. Tritium, formed when neutrons are captured by Li-6, was assumed to be removed along with xenon by sparging the fuel salt with helium.

The fertile stream is processed to remove only uranium and volatile fission products. Non-volatile fission products accumulate indefinitely, and, except for Sm-149 and Sm-151, do not burn out completely. In the MSBR, the loss of neutrons to fission products in the fertile stream is controlled by removing U-233 rapidly so that few fissions occur in the fertile stream and by discarding the fertile stream after 20 years. The "equilibrium" loss was set equal to the average loss during this period.

6.2.2 MSBR Key Variables

The temperature of the fuel-stream entering the heat exchanger was fixed conservatively at 1300°F to avoid excessive corrosion and mass transfer of chromium, which tends to be leached from INOR by hot salt and to be deposited, sometimes as a scale and sometimes intergranularly in the alloy, from colder salt. The temperature difference between the entering fuel stream and the outlet temperature of the primary coolant is also limited by considerations of thermal stress in the tube walls and tube sheets.

The fuel-stream outlet temperature (1125°F) was selected by a consideration of the heat transfer and pumping power requirements. The viscosity of the carrier salt ($37\% \text{BeF}_2$) increases rapidly at lower temperatures. Further, in order to be reasonably certain that the fuel salt will not be inadvertently frozen, the temperature of the primary coolant entering the heat exchanger should be appreciably greater (say 100°F) than the liquidus temperature of the fuel stream (850°F). Since the temperature of the fuel in thermal contact with coolant at the inlet must be 50 to 100 degrees higher than the coolant temperature, an outlet fuel temperature not less than 1050°F is indicated on this basis.

Among the design bases discussed in Sec. 3 was the requirement that not fewer than two reactors should be employed in a single power station. This, taken with the overall station efficiency of 42.3, requires only 1182 Mwt from each of two molten-salt reactors. The diameter of the core and the volume fraction of fuel, when selected to optimize the nuclear performance, as discussed below, result in a reactor in which the power requirements are easily satisfied with average fuel velocities of 24 fps. The ratio of peak-to-average radial power is about 1.7, and the corresponding peak velocity is about 40 fps. This can be reduced, however, by distributing the fuel channels in ways that flatten the distribution of the fission rate, as discussed in Sec. 5.5.5. At this velocity, the tensile stress imposed on the ~~outer graphite thimble of the fuel tubes is only ~ 63 psi, well below the~~ maximum allowable stress of 1000 psi.

The thickness of the blanket in the MSBR is related to the performance mainly through the loss of neutrons by leakage and to capture in protactinium. At moderate to short fertile-stream cycle times, the concentration of U-233 in the fuel stream is quite low and very few fissions occur in the fertile stream ($\sim 2\%$). As a result, the leakage is small (~ 0.005 neutrons per neutron absorbed in fissile atoms) with blankets 2 ft thick. However, the method specified for processing the fertile stream of the MSBR (fluoride volatility) does not remove protactinium, and it is necessary to increase the thorium inventory to control the loss of neutrons to protactinium. With fertile-stream salt containing 15 mole % ThF_4 , a blanket 3 ft thick requires

an inventory of ~ 140 tonnes of thorium. Leakage from such blankets is very low (~ 0.002 neutrons), but losses to protactinium are still appreciable (~ 0.01), and it is desirable to increase the thorium inventory further. Since there is little to be gained in the way of reduced leakage by putting additional thorium inside the reactor, and a larger, more expensive vessel would be required, additional fertile-stream volume can conveniently be stored in dump-tanks. This stored fluid, however, must be mixed with the circulating fluid frequently if the expected benefit is to be realized.

In a series of preliminary calculations, the volume fraction of fuel stream in the core, the volume fraction of the fertile stream, the core diameter, and the fuel concentration in the fuel stream were adjusted to optimize the nuclear performance with respect to mean η of fuel isotopes and parasitic absorptions in moderator and in fuel carrier, while simultaneously satisfying the criticality and power removal requirements at the limiting conditions of temperature and flow. The performance, as measured by $\bar{\eta} - A(\text{mod}) - A(\text{salt})$, exhibits a broad maximum at carbon-uranium ratios in the range from 5000 to 6000 with fuel-stream volume fractions in the range 0.15 to 0.20, fertile stream volume fractions of 0.06 to 0.07, and core diameters of 7 to 8 ft. A consistent set of specifications, given in Table 5.2.1 were selected, and parametric studies of the remaining (key) variables were made.

These key variables are, in addition to the thorium inventory mentioned above, the fertile-stream processing cycle time, the fuel-stream processing time, and the fuel-stream (barren-salt) discard cycle time. The optimum combination of the three cycle times was determined as a function of the fuel yield as described in Appendix E.

6.2.3 MSBR Results

The optimum relation between fuel cycle cost and fuel yield was determined by parametric studies of the key variables listed in Sec. 6.2.2. In two series of preliminary calculations, the effect of varying the thorium inventory was studied. Then, for a selected thorium inventory, the fuel-cycle time, the fuel-salt discard cycle time, and the fertile stream processing

cycle time were optimized with respect to fuel cost at various fuel yields. The MSBR is capable of breeding over a wide range of operating conditions, with fuel costs ranging from about 0.6 mills/kwhr up to 1.5 mills/kwhr and corresponding fuel yields ranging from zero up to about 7% per year, as illustrated in Fig. 6.2.1.

The relation of thorium inventory to fuel yield and fuel cycle cost was studied in a representative case having a fertile stream processing cycle time of 35 days, a fuel-stream cycle time of 20 days, and a fuel-salt discard cycle time of 1000 days was selected for study. The thorium inventory in the fertile stream was varied progressively from 100 to 400 tonnes by increasing the fertile stream volume from 2000 to 9000 ft³. This particular series of calculations was performed at an early stage in the study, and the combination of cycle times was not optimum with respect to fuel cycle cost at the various fuel yields. However, the dependency of fuel yield and cost on thorium inventory, as will be shown later, is only weakly affected, if at all, by choice of cycle times. Therefore, the behavior exhibited by the selected case is typical, and may be used as a guide in selecting thorium inventories. ~~The important results from these calculations are given~~ below.

Molten-Salt Breeder Reactor

Dependency of Fuel Yield and Fuel Cycle Cost on
Thorium Inventory

Thorium Inventory	Relative Fuel Yield	Relative Fuel Cycle Cost
100	1.0	1.00
140	1.2	1.03
180	1.3	1.09
270	1.4	1.20
400	1.4	1.39

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Fig. 6.2.1. MSBR Performance.

As the thorium inventory is increased, losses to protactinium decrease, and there are gains in respect to mean eta and U-236 absorptions. Breeding gain increases, but at a decreasing rate. Meanwhile, fuel inventory in the fertile stream rises. As a result, fuel yield rises, rapidly at first, and then slowly as the influence of inventory overrides that of breeding gain. The cost increases steadily, however, being driven up by increased charges for thorium and uranium. The fuel yield reaches a point of negligible improvement at 270 tonnes of thorium. This inventory was used in further studies.

As mentioned in a previous section, 100 tonnes of thorium will not fill the blankets when these are three ft thick. In the corresponding calculation, no adjustment was made for the greater leakage that would result if 100 tonnes were used. Thus, for that case in the table, the fuel yield should be less and the cost higher than calculated. The 140 tonne case was therefore selected as a representative low-thorium case for further study.

The dependency of fuel yield and fuel cycle cost on fuel stream cycle time was studied. A comparison between two series of calculations is shown below.

Molten-Salt Breeder Reactor

Dependency of Fuel Yield and Fuel Cycle Cost on Thorium Inventory with 50-Day Fertile-Stream Processing and Various Fuel-Stream Processing Cycle Times

Thorium Inventory, tonnes					
270	140	Diff.	270	140	Diff.
Fuel Yield, %/yr			Fuel Cost, mills/kwhr		
2.0	0.3	1.7	0.85	0.63	0.22
2.5	0.7	1.8	0.84	0.63	0.21
3.6	1.7	1.9	0.87	0.66	0.21
4.6	2.7	1.9	0.95	0.75	0.20
5.6	2.7	1.9	0.95	0.75	0.20
6.6	4.7	1.9	1.57	1.37	0.20

The fuel-stream processing cycle time and the fuel-salt discard cycle time were optimized with respect to fuel cycle cost at each fuel yield studied. The fertile stream cycle time was 50 days in both series. In one series, the thorium inventory was 140 tonnes, in the other, 270 tonnes. Although some slight trends with increasing fuel-stream cycle times are observed, the rule can be formulated that doubling the thorium inventory adds about 1.9% per year to the fuel yield and about 0.2 mills/kwhr to the fuel cycle costs regardless of the fuel-stream cycle times. Although not supported by calculation, the assumption was made that this rule could be extended to other cycle times for both fuel and fertile stream processing.

In an extensive series of calculations, cycle times for processing of fuel and fertile streams and the discard of fuel-salt were optimized with respect to fuel cycle cost at fuel yields ranging from 2 to 7% per year, as described in Appendix E. Details of performance of optimized reactors (cases 1 through 7) are given in Table 6.2.1, etc. The thorium inventory was 270 tonnes throughout, the fertile stream cycle time ranged from 35 to 200 days, the fuel stream cycle time ranged from 12 to 84 days, and the fuel-salt discard cycle time ranged from 145 to 1550 days. As the cycle times increase, losses to protactinium fall about 10% because the Pa concentration falls about 7% (due to falling breeding ratio), and also because the volume fraction of fertile stream in the core falls about 7% (60% of the captures in protactinium occur in the core).

Concurrently captures by samarium and other fission products rise by ~ 0.04 neutrons. This is more than half the greatest breeding gain, and cuts the production of excess fuel in half. As a result, the rate of purging of higher isotopes of uranium from the system decreases. The increase in U-235 concentration causes a decline in mean effective η of 0.004 and the U-236 captures increase by 0.007.

A breakdown of inventories is given in Table 6.2.2. It is seen that as the cycle times increase, the principal change is in the inventory of fissile isotopes in the fertile stream, which rises by a factor of five. This increase in the fertile stream makes possible a slight decrease in the fissile inventory in the fuel stream. The net effect, however, is a 40% increase in

Table 6.2.1. MSBR Nuclear Characteristics
Station Power: 1000 Mwe Fuel Processing: Continuous, on-site

	Case Number							
	1	2	3	4	5	6	7	8
	Key Variables							
Moderator-fuel ratio, N(C)/N(U)	←----- 5100 -----→							
Volume fraction of fuel-stream in core	←----- 0.16 -----→							
Volume fraction of fertile-stream in core	0.071	0.070	0.069	0.068	0.068	0.066	0.065	0.066
Thorium inventory, tonnes	←----- 270 -----→							
Fertile-stream cycle time, days	35.0	35.0	35.0	50.0	100	200	200	50.0
Fuel-stream cycle time, days	11.0	23.0	31.0	55.0	55.0	72.0	84.0	50.0
Fuel-salt discard cycle time, days	145	190	235	400	400	770	1550	450
	Nuclear Economy							
Net neutron yield, $\bar{\eta}$	2.2137	2.2131	2.2128	2.2120	2.2115	2.2103	2.2097	2.2095
Nuclear losses ^a to:								
U-233 absorptions	0.9172	0.9150	0.9139	0.9111	0.9095	0.9053	0.9034	0.9021
U-235 absorptions	0.0828	0.0850	0.0861	0.0889	0.0905	0.0947	0.0966	0.0979
Th-232 fissions	0.0019	0.0019	0.0018	0.0018	0.0018	0.0018	0.0018	0.0018
Pa-233 absorptions (x 2)	0.0120	0.0117	0.0116	0.0113	0.0112	0.0108	0.0106	0.0214
U-236 absorptions	0.0106	0.0115	0.0119	0.0132	0.0140	0.0164	0.0176	0.0165
Np-237 absorptions	0.0004	0.0010	0.0010	0.0008	0.0008	0.0010	0.0011	0.0010
Xe absorptions	0.0050	0.0050	0.0050	0.0050	0.0050	0.0050	0.0050	0.0050
Sm absorptions	0.0001	0.0001	0.0001	0.0002	0.0003	0.0005	0.0005	0.0003
Other fission product absorptions	0.0218	0.0267	0.0317	0.0422	0.0442	0.0532	0.0631	0.0478
Corrosion product absorptions	0.0008	0.0008	0.0008	0.0008	0.0008	0.0008	0.0008	0.0008
Core moderator absorptions	0.0291	0.0286	0.0286	0.0287	0.0287	0.0287	0.0288	0.0287
Fuel carrier absorptions	0.0284	0.0302	0.0302	0.0302	0.0291	0.0301	0.0301	0.0301
Thorium carrier absorptions	0.0177	0.0199	0.0198	0.0196	0.0195	0.0192	0.0190	0.0192
Loss of delayed neutrons	0.0043	0.0043	0.0043	0.0043	0.0043	0.0043	0.0043	0.0043
Leakage	0.0016	0.0016	0.0016	0.0016	0.0016	0.0016	0.0016	0.0016
Chemical processing losses	0.0022	0.0022	0.0022	0.0022	0.0022	0.0022	0.0022	0.0022
Net breeding gain	0.0778	0.0676	0.0622	0.0501	0.0480	0.0347	0.0232	0.0288

Table 6.2.1. (Continued)

	Case Number							
	1	2	3	4	5	6	7	8
	Other Characteristics							
$\xi \Sigma_s / N(U)$, barns/atom (U-233 + U-235)				3900				
Per cent thermal fissions				65-75				
Per cent fissions in fertile stream	1	1	1	2	3	6	6	3

a. Neutrons or fissile atoms lost per neutron absorbed in fuel atoms.

Table 6.2.2. MSBR Inventories and Processing Rates

	Case Number							
	1	2	3	4	5	6	7	8
Key Variables								
Fertile stream cycle time, days	35.0	35.0	35.0	50.0	100	200	200	50.0
Fuel stream cycle time ^a , days	11.0	23.0	31.0	55.0	55.0	72.0	84.0	50.0
Fuel salt discard cycle time ^a , days	145	190	235	400	400	770	1550	450
Station-Inventories								
Thorium, tonnes	←				270	→		140
Fuel carrier (fuel excluded), tonnes	←				28	→		
Thorium carrier (ThF ₄ excluded), tonnes	←				200	→		100
Fissile inventory, kg								
U-233 in fuel stream	470	469	469	465	457	441	441	453
U-235 in fuel stream	46	48	48	50	51	53	55	55
U-233 in fuel-stream processing	26	10	10	15	14	14	14	14
U-235 in fuel-stream processing	3	1	1	2	2	2	2	2
U-233 in fuel reserve	68	68	68	67	66	63	63	65
U-235 in fuel reserve	7	7	7	7	7	8	8	8
U-233 + U-235 in fuel dump tanks	52	52	52	51	51	49	50	51
U-233 in fertile stream	95	94	93	131	266	489	482	125
Pa-233 in fertile stream	108	107	107	105	105	103	102	102
U-233 in fertile-stream processing	3	3	3	3	3	2	2	2
Total, kg per station	878	859	858	896	1022	1224	1219	877
Processing Rates								
Spent fuel processing rate, kg(U-233+U-235)/day	28.4	11.4	11.4	16.2	15.9	15.5	15.5	16.0
Excess fissile uranium produced, kg/day	0.206	0.187	0.172	0.139	0.139	0.095	0.064	0.081
Thorium processing rate, kg/day	7510	7510	7510	5250	2630	1310	1310	2630
Thorium replacement rate ^b , kg/day	39.8	39.7	39.7	39.4	39.0	38.8	38.7	21.0
Fuel salt replacement rate ^a , kg/day	194	148	120	67.0	57.5	36.6	18.2	62.6
Fertile stream loading, gm(U-233+Pa-233)/kg Th	0.36	0.36	0.36	0.50	1.0	1.9	1.9	1.0

a. Cycle time optimized to give minimum fuel cycle cost.

b. Based on a 20-year discard cycle to remove fission products. Also includes thorium burnup.

the total inventory. This, combined with the decline in breeding gain causes the fuel yield to fall sharply from 6.9 to 1.6, as indicated in Table 6.2.3.

The greatest fuel yield (6.9%/yr) was calculated for case 1. An examination of the neutron losses listed in Table 6.3.1 does not disclose any that could be significantly reduced by further optimization. Efforts to increase the neutron yield, $\bar{\eta}_e$, would be more than offset by increases in losses to moderator and fuel carrier. The losses to fuel carrier could be reduced by decreasing the volume fraction of the fuel stream in the core, but this would drive the inventory of fuel in the external fuel system up sharply, and give no increase in the fuel yield. Increasing the fuel processing rate without limit would not decrease the loss to "other fission products" below 0.017, corresponding to the loss to isotopes soluble in HF solutions and which are removed from the salt stream only by discarding the salt. The loss to Pa of 0.01 corresponds to a volume of fertile stream of 6000 ft³ per station, of which only half is inside the reactor vessels. It is not practical or economical to reduce the loss to Pa by further increases in the volume of the fertile stream. It seems clear that breeding gains in excess of 0.08 and fuel yields in excess of 7%/yr cannot be achieved in the MSBR.

Fuel cycle costs for the MSBR are broken down in Table 6.2.3. As the cycle times increase, inventory charges on fissile isotopes increase, and breeding credit falls by a factor of three. These costs are more than compensated, however, by decreased processing plant costs and fuel-salt discard cost. The net cost falls from 1.6 to 0.7 mills/kwhr. The corresponding fuel yields are 6.9 and 2.7% per year.

The data for cases 2 through 7 from Table 6.2.3 are plotted in Fig. 6.2.1, and lie near an envelope of performance curves for stations having thorium inventories of 270 tonnes. In the table on p. 148, the 140 tonne case having a fuel yield of 2.7% corresponds to case 4 in Table 6.2.3, and it is plotted in Fig. 6.2.1, also. The dashed curve through this single point, while unsupported by calculation, is plausibly constructed by use of the rule derived from the table. It is clear that a large number of such curves, each corresponding to a different thorium inventory, would define a locus of maximum performance for molten-salt reactors, and that, by making

Table 6.2.3. MSBR Fuel Costs and Performance Summary

	Case Number							
	1	2	3	4	5	6	7	8
	Key Variables							
Thorium inventory, tonnes	← 270 → 140							
Fertile-stream cycle time, days	35.0	35.0	35.0	50.0	100	200	200	50.0
Fuel-stream cycle time ^a , days	11.0	23.0	31.0	55.0	55.0	80.0	84.0	50.0
Fuel salt discard cycle time ^a , days	145	190	235	400	400	655	1550	450
	Cost Breakdown, mills/kwhr							
Fuel cycle costs ^{b,c} , mills/kwhr								
Uranium inventory	0.075	0.074	0.074	0.078	0.087	0.104	0.104	0.075
Thorium inventory	0.132	0.132	0.130	0.130	0.130	0.129	0.129	0.066
Fuel carrier inventory	0.020	0.022	0.020	0.020	0.020	0.021	0.021	0.020
Thorium carrier inventory	0.166	0.167	0.167	0.165	0.163	0.160	0.160	0.082
Fuel-stream processing plant	0.628	0.430	0.365	0.255	0.255	0.239	0.204	0.267
Fertile-stream processing plant	0.261	0.261	0.261	0.209	0.128	0.090	0.091	0.138
Thorium replacement	0.059	0.027	0.026	0.027	0.027	0.026	0.026	0.013
Fuel carrier replacement	0.404	0.329	0.244	0.143	0.143	0.077	0.038	0.130
Thorium carrier replacement	0.034	0.034	0.034	0.034	0.033	0.033	0.033	0.017
Breeding credit	0.129	0.117	0.107	0.087	0.084	0.076	0.039	0.050
	Performance Summary							
Net fuel-cycle cost, mills/kwhr	1.65	1.36	1.21	0.97	0.90	0.80	0.77	0.76
Fuel yield ^c , %/yr	6.9	6.4	5.9	4.5	3.9	2.3	1.6	2.7
Fuel specific power, Mwe/kg	1.12	1.17	1.17	1.12	0.98	0.82	0.82	1.14
Thorium specific power, Mwe/tonne	3.7	3.7	3.7	3.7	3.8	3.8	3.8	7.5
Net breeding gain (rounded)	0.08	0.07	0.06	0.05	0.05	0.03	0.02	0.03
Per cent of heat from fertile stream	10	10	10	10	12	14	14	11
Poison fraction ^d	0.022	0.027	0.032	0.042	0.045	0.054	0.064	0.048

a. Cycle time optimized to give minimum fuel cycle cost.

b. See Sec. 3.2 for cost bases.

c. Based on 80% plant factor.

d. Neutrons lost to fission products per neutron absorbed in fissile atom. Poison fraction equals sum of losses to Sm and other fission products (except Xe) in both fuel and fertile streams.

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a small allowance for further possible optimizations in moderator uranium ratios, etc., the locus would lie about as indicated in the figure. From this estimated locus, it was concluded that the MSBR is capable of fuel yields of about 7% per year at fuel cycle costs not greater than 1.5 mills/kwhr, and that fuel cycle costs as low as 0.6 mill/kwhr can be achieved in self-sustaining systems.

6.3 Liquid Bismuth Breeder Reactor (LBBR)

6.3.1 LBBR Special Assumptions

In the LBBR, the necessity of using parallel, circular, fuel channels dictates a cylindrical geometry for the core. The heat obtainable from the core is determined by its diameter, volume fraction of fuel-stream, velocity of fuel, and temperature rise. The fissile inventory is proportional to the volume, the volume fraction of fuel-stream, and the concentration of the fuel in the fuel stream. Thus, the specific power (Mwt/kg of fuel) tends to decrease as the length of the core is made greater than the diameter. If the length is made less than the diameter (pancake core), the fuel concentration required for criticality eventually rises and the inventory in the external circuit rises, bringing the power density down. The optimum ratio of length to diameter cannot be far from unity. The nuclear characteristics of such cores can be estimated satisfactorily by means of one-dimensional calculations on "equivalent" spherical cores. Equivalence is defined to exist when the bucklings of the bare spherical core is made equal to that of the bare cylindrical core.

The composition of the end-blankets, through which the fuel stream passes, differs appreciably from that of the radial blanket. The "equivalent" blanket for the spherical core was formulated by taking a weighted average of compositions of the blankets of the cylindrical core. The weighting factors were proportional to the areas of the inner surfaces of the end and side blankets.

The heterogeneity of the fuel and moderator was treated by the use of disadvantage factors as described in Sec. 3 and Appendix E. The heterogeneity of the fertile stream and the moderator was ignored, and these were treated as a homogeneous mixture. This procedure probably resulted in an overestimate of the effective group cross sections of thorium in the resonance region of neutron energies, but the error can be interpreted as a slight underestimate of the fertile-stream volume fraction. The underestimate in the volume fraction only slightly affects the estimate of the thorium inventory.

The analysis by Thomas, et al (22) of the parasitic captures in bismuth in the LMFR was adopted for the present study. Agreement was achieved in the LBBR calculations by taking a thermal cross section. They took into account saturation and self-shielding in the two known resonances at 790 and 2290 ev, which had been studied in bismuth-graphite lattices by Rothenstein and Chernenick (134). Agreement in the LBBR calculations was achieved by taking an effective thermal cross section that gave the same total absorptions estimated by Thomas at specified ratios of bismuth to graphite. The effective cross section was found to be 0.026 barns, compared to a value of 0.017 barns at 1200°F estimated by Nestor (24).

6.3.2 LBBR Key Variables

Seven of the most important degrees of freedom in the design of cores for the LBBR are listed below:

1. Fuel-stream processing cycle time
2. Volume fraction of fuel stream in core
3. Concentration of fuel atoms in fuel stream
4. Dimensions of core
5. Ratio of moderator atoms to fertile atoms
6. Fertile stream processing cycle time
7. Thorium inventory.

Of minor importance are the fuel channel dimensions and lattice spacing, fertile-stream channel shape, dimensions, and spacing, thickness of blanket, and volume fraction of moderator in blanket.

In optimizing the LBBR, any coupling among the important variables must be considered. For example, the volume fraction of fuel in the core and the core diameter must be compatible with the power requirement, the maximum allowable velocity of fuel in the core, and the permissible temperature rise. Again, at a given diameter, and volume fraction of fuel, the concentration of fuel atoms in the fuel stream must satisfy the criticality condition without approaching the solubility limit too closely.

The important dependent variables must be recognized and their relation to the independent variables determined. These include the effective η of the system and the fraction of neutrons absorbed by the moderator (both of which increase with increasing moderator-fuel atom ratio), parasitic captures in fuel carrier (which depend in part on the volume fraction of fuel in the core), and inventory of fissile isotopes in the fuel system (which depends mainly on the fuel-atom concentration in the fuel stream). The presence of thorium in the core of the LBBR provides an additional degree of freedom in optimizing the performance. For instance, the diameter of the core can be partially decoupled from the moderator-fuel ratio in the critical equation by adding thorium.

Fortunately, the range of optimum and compatible values of several of these variables were narrowed by Thomas, et al in their detailed study (22). In the calculations reported here, the reactor specifications recommended by Thomas, et al, were selected as a starting point. Adjustments were made to satisfy the slightly different power requirements, and the resulting system was evaluated in respect to the important dependent variables listed above. ~~Compatible changes in the independent variables which showed promise of improving the performance were then made and the system was reevaluated.~~ In this way, a near-optimum set of compatible specifications was obtained.

Thomas (22) reports that a fuel-stream processing cycle time of about 12 days is sufficient to hold the loss of neutrons to fission products (other than xenon) to 0.02 neutrons per neutron absorbed in fissile atoms. This loss is less than the corresponding loss in other systems studied, and may prove difficult to achieve if the efficiency of the extraction plant is less than estimated. The associated cost (~ 0.1 mills/kwhr), though appreciable, is not large, and savings at longer cycle times would be offset considerably by decreased breeding credit. Also, since the fission products are removed in several groups by several different methods, it is not easy to relate the fission product poison fraction to the processing rate or the rate to the cost. For these reasons, further optimization of the fuel-stream processing cycle time was not attempted.

Thomas further reports that the optimum thorium inventory and processing cycle time are such that the product stream contains about 6 grams of bred material (U-233 and Pa-233) per kg of thorium. Accordingly, in the first series of calculations reported here, the thorium inventory and cycle time were varied proportionately to maintain a loading of 6 grams per kg.

By means of preliminary calculations not reported here it was found that blankets containing 50% graphite by volume should be at least four ft thick. The leakage then amounts to less than 0.005 neutrons per neutron absorbed in fissile atoms, and is satisfactorily low. About 60 tonnes of thorium (as ThO_2) per station is required to fill these blankets. If additional thorium is used to reduce losses to Pa-233, it may be stored in tanks outside the reactor, but it must be mixed with the circulating blanket fluid inside the reactor at least once a month, and preferably once a week when the cycle time for processing the fertile stream is as short as 100 days.

In other preliminary calculations, the atom ratio of moderator to fuel was progressively increased. The difference between effective eta and the parasitic captures in graphite was found to pass through a broad maximum in the atom ratio range of 5000 to 7000.

Parasitic captures in bismuth were reduced by progressively reducing the volume fraction of fuel in the core. This was done by reducing the diameter of the fuel channels and altering the lattice spacing in order to maintain the moderator-fuel ratio constant. Since the fuel velocity and temperature rise were fixed at the maximum allowable, it was necessary progressively to increase the core diameter in order that more fuel channels would be available for heat generation. Criticality was maintained by adjusting the volume fraction of fertile stream in the core. The captures in bismuth passed through a minimum at a fuel-stream volume fraction of about 0.2.

It was possible to optimize the variables discussed so far with respect to the fuel yield and without regard to fuel cycle cost because their influence on the cost is small. The strongest connection is through the inventory of fissile atoms in the fuel stream. But the charge on the total plant inventory is only about 15% of the fuel cycle cost, and less than half the total inventory is contained in the fuel-stream.

The remaining (key) variables, which must be optimized with respect to the fuel cycle cost at a given fuel yield, are the thorium inventory and the processing cycle time.

6.3.3 LBRR Results

Nuclear Performance. - Detailed results for nine cases are given in Table 6.3.1, etc. Cases 1 through 5 comprise a series in which the thorium inventory and fuel cycle time were increased progressively in such a manner that the fertile-stream loading (grams of U-233 plus Pa-233 per kg of thorium) remained constant at about 6 g/kg. The net breeding gain increased slowly from 0.024 to 0.057. This improvement was due almost entirely to a reduction in losses to protactinium, which fell from 0.043 to 0.018 as the thorium inventory was tripled. There was also a slight improvement in η due to reduction in the number of atoms passing through the uranium-isotope chain to U-236.

The inventory of fissile isotopes also increased, as indicated in Table 6.3.2. The increase was entirely in the fertile stream, and was proportional to the increase in thorium. Its effect eventually overcame the rise in the breeding gain, and the fuel yield passed through a broad maximum (Table 6.3.3).

Cases 8, 2, and 9 constitute a series in which the thorium inventory ~~was held constant at about 120 tonnes while the fertile-stream processing~~ cycle time was increased from 195 to 475 days. This caused the fertile stream loading to increase from 4 to 8 grams per kg of thorium while the breeding gain fell from 0.052 to 0.034.

This decline was the result of several factors. At the high loading, the rate of fissions in the fertile stream increased, resulting in the accumulation of more fission products and consequently greater parasitic captures. The increase (from 0.025 to 0.037) was not compensated by small decreases in losses to graphite and fuel- and fertile-stream carriers (Bi). When the breeding gain decreased, sales fell off, and the rate of removal of U-235 and U-236 via sales decreased. The U-235 accumulated to higher levels, causing a decrease in the mean η of the system. As the loading rises, the fissile

Table 6.3.1. LBBR Nuclear Characteristics

Station Power: 1000 Mwe Fuel Processing: Continuous, on-site

	Case Number								
	1	2	3	4	5	6	7	8	9
Key Variables									
Moderator-fuel ratio, N(C)/N(U)	5600	5600	5600	5600	5600	6600	6800	5500	5600
Volume fraction of fuel-stream in core	0.20	0.20	0.20	0.20	0.20	0.20	0.15	0.20	0.20
Volume fraction of fertile-stream in core	0.068	0.071	0.072	0.073	0.074	0.060	0.077	0.072	0.059
Core diameter, ft	6.4	6.4	6.4	6.4	6.4	6.4	7.4	6.4	6.4
Station thorium inventory, tonnes	61	90	119	148	177	119	119	116	116
Fertile stream processing cycle time, days	161	242	322	403	484	329	316	195	475
Neutron Economy									
Net neutron yield, $\bar{\eta}$	2.2038	2.2060	2.2069	2.2074	2.2077	2.2125	2.2149	2.2059	2.2018
Nuclear losses									
U-233 absorptions	0.8901	0.8983	0.9020	0.9040	0.9052	0.9011	0.9055	0.9044	0.8967
U-235 absorptions	0.1099	0.1017	0.0980	0.0960	0.0948	0.0989	0.0945	0.0956	0.1033
Th-232 fissions	0.0024	0.0025	0.0025	0.0025	0.0025	0.0024	0.0023	0.0024	0.0028
Pa-233 absorptions (x 2)	0.0430	0.0319	0.0253	0.0209	0.0178	0.0248	0.0244	0.0226	0.0244
U-236 absorptions	0.0192	0.0165	0.0156	0.0152	0.0151	0.0160	0.0138	0.0148	0.0187
Np-237 absorptions	0.0003	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002
Xenon absorptions	0.0104	0.0103	0.0101	0.0097	0.0094	0.0105	0.0104	0.0082	0.0113
Other fission product absorptions ^b	0.0185	0.0191	0.0195	0.0197	0.0199	0.0201	0.0197	0.0175	0.0264
Core moderator absorptions	0.0256	0.0255	0.0255	0.0255	0.0254	0.0320	0.0350	0.0263	0.0244
Fuel carrier absorptions ^c	0.0253	0.0253	0.0253	0.0253	0.0253	0.0318	0.0244	0.0273	0.0249
Thorium carrier absorptions ^d	0.0219	0.0223	0.0224	0.0225	0.0226	0.0251	0.0244	0.0239	0.0216
Loss of delayed neutrons	0.0047	0.0047	0.0047	0.0047	0.0047	0.0047	0.0047	0.0047	0.0047
Leakage	0.0040	0.0041	0.0041	0.0041	0.0041	0.0039	0.0039	0.0031	0.0058
Allowance for additional end-losses	0.0010	0.0010	0.0010	0.0010	0.0010	0.0010	0.0010	0.0010	0.0010
Processing losses	0.0033	0.0032	0.0032	0.0032	0.0032	0.0032	0.0032	0.0032	0.0032
Net breeding gain	0.0242	0.0394	0.0475	0.0529	0.0565	0.0367	0.0475	0.0507	0.0324
Other Characteristics									
$\Sigma_s/N(U)$, barns/atom (U-233+U-235)	4200	4200	4200	4200	4200	5000	5200	4100	4200
Per cent thermal fissions	71	71	71	71	71	75	76	70	69
Per cent fissions in fertile stream	15	17	18	18	19	19	18	12	24

a. Neutrons or fissile atoms lost per neutron absorbed in fissile atom.

b. Including samarium.

c. No allowance for corrosion products.

d. Includes blanket moderator and Bi.

Table 6.3.2. LBBR Station Inventories and Processing Rates

	Case Number								
	1	2	3	4	5	6	7	8	9
	Key Variables								
Station thorium inventory, tonnes	61	90	119	148	177	119	119	116	116
Fertile stream processing cycle time, days	161	242	322	403	484	329	316	195	475
	Station Inventories								
Station bismuth inventory, tonnes	820	1064	1304	1544	1784	1304	1321	1307	1300
Station fuel inventory, kg									
U-233 in fuel stream ^a	703	695	689	685	682	571	796	707	580
U-235 in fuel stream ^a	113	104	101	99	98	84	110	92	98
U-233 in fuel-stream processing	59	58	58	58	58	48	64	59	54
U-235 in fuel-stream processing	10	9	8	8	8	7	9	8	9
U-233 in fuel-stream reserve	53	54	53	52	52	52	53	58	48
U-235 in fuel-stream reserve	9	8	8	8	8	8	7	8	8
U-233 in fertile stream	264	432	604	776	948	598	586	374	828
Pa-233 in fertile stream	81	88	93	96	98	92	92	88	94
U-233 in fertile-stream processing	78	52	54	58	60	53	54	75	41
Pa-233 in fertile-stream processing	24	16	12	10	8	11	13	18	9
Total, kg per station	1394	1516	1680	1850	2020	1524	1784	1487	1769
	Processing Rates								
Spent fuel processing rate ^b , kg/day	68.6	67.2	66.4	65.9	65.5	55.1	36.3	66.7	62.8
Excess fissile uranium produced, kg/day	0.0544	0.088	0.106	0.118	0.126	0.082	0.105	0.113	0.072
Thorium processing rate, kg/day	358	358	358	358	358	351	366	591	243
Thorium replacement rate, kg/day	9.26	9.61	9.54	9.47	9.47	9.40	9.90	14.5	7.12
Bismuth replacement rate, kg/day	28.8	40.6	46.4	55.0	63.4	46.4	46.8	46.4	45.3
Fertile stream loading, g(U-233+Pa)/kg Th	6.0	6.0	6.0	6.0	6.0	6.0	5.9	4.0	8.0

a. Includes allowance for heels in dump tanks.

b. U-233 and U-235 only.

Table 6.3.3. LEBR Fuel Costs and Performance Summary

	Case Number								
	1	2	3	4	5	6	7	8	9
	Key Variables								
Station thorium inventory, tonnes	61	90	119	148	177	119	119	121	118
Fertile stream processing cycle time, days	161	242	322	403	484	329	316	195	475
	Cost Breakdown ^a , mills/kwhr								
Uranium inventory	0.119	0.130	0.144	0.158	0.171	0.130	0.153	0.127	0.152
Thorium inventory	0.030	0.044	0.058	0.073	0.087	0.058	0.058	0.059	0.058
Bismuth inventory	0.074	0.095	0.117	0.139	0.160	0.117	0.119	0.117	0.114
Fuel stream processing	0.103	0.103	0.103	0.103	0.103	0.103	0.103	0.103	0.103
Fertile stream processing	0.774	0.774	0.774	0.774	0.774	0.774	0.786	0.898	0.699
Thorium replacement	0.013	0.014	0.013	0.013	0.013	0.013	0.014	0.020	0.010
Bismuth replacement	0.008	0.010	0.012	0.014	0.016	0.012	0.012	0.012	0.012
Breeding credit ^b	0.042	0.069	0.083	0.092	0.098	0.064	0.082	0.088	0.057
	Performance Summary								
Net fuel cycle cost ^b , mills/kwhr	1.08	1.10	1.14	1.18	1.23	1.14	1.16	1.24	1.09
Fuel yield ^b , %/year	1.43	2.12	2.32	2.34	2.30	1.97	2.17	2.80	1.50
Fuel specific power, Mwe/kg	0.72	0.66	0.60	0.54	0.50	0.66	0.56	0.67	0.57
Thorium specific power, Mwe/tonne	16.4	11.1	8.4	6.7	5.7	8.4	8.4	8.3	8.5
Net breeding gain (rounded)	0.024	0.039	0.048	0.053	0.057	0.037	0.048	0.051	0.032
Per cent of heat from fertile stream	21.6	23	23	25	23	25	24	18	30

a. See Sec. 3.2 for cost bases. No allowance for waste disposal.

b. Plant factor: 80%.

inventory in the blanket also rises. With breeding gain falling and inventory rising, the fuel yield falls rapidly from 3.1 to 1.7% per year (Table 6.3.2).

The largest breeding gain calculated was that of case 5. The parasitic losses are listed in Table 6.3.1. As shown below, it is uneconomic to reduce the Pa losses to 0.018 by inflating the thorium inventory, as was done in case 5. Losses to graphite cannot be reduced without incurring some loss in eta as the spectrum of neutrons hardens. As indicated in a previous section, losses to bismuth in the fuel carrier have been reduced to a minimum, consistent with the power requirements and the solubility limits of uranium in bismuth. An increase in ThO_2 concentration in the fertile-stream slurry would help reduce losses to the fertile stream carrier, perhaps by 0.01 units in slurries containing 25 wt % ThO_2 , but the physical properties of such slurries are not known.

Some saving might be made in losses to xenon and other fission products. Thomas, et al (22) report experiments which indicate that a large fraction (perhaps more than 3/4) of the fission fragments (especially xenon) are not trapped in the particles of ThO_2 but appear in the bismuth from which they ~~could conceivably be removed by a suitable process.~~ Optimistically, perhaps 0.004 units could be saved by removing xenon, and 0.010 units by removing other fission products, optimizing eta, etc.

Such savings might raise the breeding gain by 0.024. They would not, however, affect the inventories, and the fuel yield in case 5 would not be increased above 3.3% per year, which corresponds to a doubling time of 30 years. Efforts to improve the performance of the LBBR must therefore be focused on reducing the inventory of fissile isotopes.

An examination of the breakdown for case 5, given in Table 6.3.2, shows that the major contributions to the inventory are those of the fuel and fertile streams. The relation of the inventory in the fuel stream to power requirement was mentioned in a previous section. Bismuth, because of its low heat capacity, is a poor coolant, and large flow rates are required. Since the velocity is limited to 33 fps, flow areas must be large,

and this leads to large volumes which result in high inventories of fissile isotopes in the fuel stream. The only hope of effecting a reduction here is to develop equipment that will handle bismuth at appreciably higher velocities.

In the fertile stream, the fissile inventory is independent of the bismuth inventory; it is tied directly to the thorium inventory and the processing cycle time. The effects of varying these together and separately were discussed above. The only hope for saving inventory here lies in developing a fertile system from which U-233 can be removed economically at a high rate. Thorex does not appear to be such a process.

A breakdown of fuel cycle costs for the LBBR is given in Table 6.3.3, together with a summary of performance characteristics. The fuel cycle costs calculated all lie below 1.5 mills/kwhr, and they range down toward 1 mill/kwhr. The costs have been plotted as a function of fuel yield in Fig. 6.3.1, and, while the points are few, the general trend is clear. Points lying on curve 1 correspond to cases 1 through 5 of Table 6.3.1, etc., in which the thorium inventory and cycle time were increased together in order to hold the fertile stream loading constant at 6 grams of U-233 plus Pa-233 per kg of thorium. The fuel yield passes through a maximum but the cost rises indefinitely as the inventories of fertile and fissile isotopes increase.

Curve 2 corresponds to the series composed of cases 8, 2, and 9, wherein the thorium inventory was 120 tonnes, and the cycle time was varied. The fuel yield and the cost both increase with decreasing cycle time, but fuel yield reaches a limit as the loss of neutrons to protactinium approaches zero.

Curves 3 are hypothetical curves not supported by calculation, but which plausibly correspond to curve 2 at higher and lower thorium inventories. It is readily seen that a large number of such curves would generate an envelope defining a locus of maximum performance for the LBBR.

As discussed above, some small improvement in fuel yield might be obtained by further optimization or by making allowance for the escape of fission products from ThO_2 and subsequent removal from the bismuth. The

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FIG. 6.3.1 L B R PERFORMANCE

Plant Factor : 0.8

Station Power : 1000 Mwe

LEGEND

A. Blanket Loading - 6 gm (Pa-233 + U-233) / kg Th

B. Thorium Inventory - 120 Tonnes; Cycle Time Varies

C. Hypothetical Curves Similar to B With different Thorium Inventories

probable magnitude of such gains was estimated. In order to make allowance for these, and other contingencies, such as the verification of a resonance integral for protactinium lower than that used here (1200 barns), or higher fission cross sections for U-233, a locus of maximum performance was drawn well to the right of and below the points calculated. On the basis of this locus, the LBBR appears to be capable of fuel yields up to 4% per year at fuel cycle costs less than 1.5 mills/kwhr, and to be able to maintain a positive breeding gain at fuel cycle costs as low as 1.0 mill per kwhr.

6.4 Graphite-Moderated Gas-cooled Breeder Reactor (GGBR)

6.4.1 GGBR Special Assumptions

The nuclear calculations were performed in cylindrical geometry by means of the GNU program for the IBM-704 (Sec. 3.2.3). The counter-current loading technique described in Sec. 5.4.3 results in nearly uniform axial distribution of fuel and fission-product concentrations averaged over a cross section of the core. The same burnup is achieved in all channels by making the residence time in a channel inversely proportional to the mean fission rate in that channel; this results in a nearly uniform distribution of concentrations in the radial direction when concentrations in adjacent channels are averaged. At a burnup of 40% of the U-233 per pass, the fuel concentration at the midplane is 0.97 times the average concentration at the ends of the core (fresh fuel entering averaged with spent fuel leaving), and the fission-product concentration at the midplane is 1.11 times the average concentration at the ends. The approximation that the concentrations of atomic species are uniform throughout the core was accordingly adopted.

This assumption is moderately optimistic. From a nuclear standpoint, the importance of a region is roughly proportional to the square of the effective flux; on that basis, the weighted fission product concentration should have been somewhat higher and the fuel concentration somewhat lower. The error in regard to fission products was largely offset by some other assumptions concerning their mean, or "composite" cross section, discussed below. Further, the heterogeneity of the fuel was ignored, and if taken into account, would have resulted in some shielding of the fission products, thus reducing their effective concentration. In regard to the fuel, the error in average concentration is not offset but enhanced by the self-shielding; however, the only effect of the approximation used is a slight underestimate of the critical mass, which is itself but a small fraction of the total station inventory (Table 6.4.2).

In the routine calculations, the ends of the reactor were treated as bare. The effect of end blankets and reflectors on breeding ratio was estimated by allocating the axial end-current to blanket absorptions and leakage

in the same proportions computed for the radial current into the blanket. This is optimistic, as provisions for coolant flow, fuel-element charging, and fertile-stream handling at the ends of the reactor will result in the loss of some neutrons. At the selected nominal core length of 20 ft, the axial end-current amounts to less than 2% of the neutrons; however, end blankets should recover 90% of these, and the end loss thus would not exceed 0.004 neutrons per neutron absorbed in fissile atoms.

Power-density peaking in the radial direction was calculated from GNU results, and lay in the range 1.7 to 1.8. Peaking in the axial direction was assumed to be 1.3, which is somewhat less than for a bare reactor (1.57), and makes allowance for the presence of end blankets and reflectors.

By very approximate methods, it was estimated that about 2 Mwt of heat per 100 Mwt of core power would be released in the blanket by degradation of gamma rays from all sources. To this was added all energy produced by fissions in the blanket, which varied from 2 to 17% of the total as the loading of bred material ranged from 4 to 10 grams per kg of thorium. The precise split between core and blanket power is thought to be not important inasmuch as adjustments can be made in the same way that adjustments are made to achieve an integral number of reactors per station, viz., by adjusting the fuel-plate thickness in the range from 0.1 to 0.25 in. in the power equation (Eq. A-11, Appendix A).

Inelastic scattering of neutrons was not taken into account in these calculations, as the error was thought to be small. However, the GNU calculations showed an appreciable "fast fission effect" due to fissions in Th-232 in the three highest energy groups. The apparent value of epsilon was 1.013, and the effect increased the breeding ratio somewhat more than 0.028 units. Calculations by means of the S_n program for IBM-704 and GNU calculations with inelastic scattering in thorium indicate that the benefit from the fast fissions was over-estimated by about 60-70%, i.e., ϵ is more nearly 1.007, and the contribution to the breeding gain probably not more than 0.017. The results presented herein are based on 60% of the effect estimated by neglecting inelastic scattering.

The calculation of the loss of neutrons to Pa-233 in the blanket was based on the assumption that the concentration of Pa is uniform throughout the blanket. This uniformity can be achieved by shifting the fertile material about in the blanket periodically as described in Appendix D for the AHBR. The shifting is done batchwise, and the identity of individual batches is preserved. This has the advantage that the time-mean concentration of U-233 in the blanket is only half the discharge concentration; as a consequence, fissions, neutron losses to fission products, and U-233 inventory in the blanket are all reduced.

6.4.2 GGBR Key Variables

The design of a core for a solid fuel, internally cooled reactor is rather more complex than the design of a fluid fuel core for the reason that, in addition to the nuclear requirements, the designer must also satisfy more restrictive thermal and mechanical requirements.

Important reactor specifications include the temperatures, pressure, flow-rate, and physical properties of the coolant, reactor length and diameter, concentrations of fuel and moderator, volume fraction of gas passages in core, thickness of blanket, thorium inventory, and rate of processing the fuel and fertile streams.

Because the efficiency of the blanket at the ends of the reactor is low (due to the presence of ducts for ingress and egress of the cooling gas and fuel elements), a requirement in regard to the length of the core is that it shall be long enough that the axial leakage will be small compared to the radial leakage. A second requirement is that it shall be long enough to provide ample volume and surface of fuel so that the heat generated in the fuel may be transferred to the gas at temperatures, stresses, and pressures not exceeding tolerable limits on the one hand, but satisfying the requirements for high thermodynamic efficiency and specific power on the other.

In regard to the volume of the coolant passages, this must be great enough to permit the removal of the heat without excessive expenditure of pump-work but not so great that the core is depleted of moderator. If the core is too porous, it must, in order to achieve criticality, be large, which

leads to low pressures through the limitation imposed by the available strength and thickness of the pressure vessel material, or else the mass of fissile atoms must be increased, which has an adverse effect on the spectrum of neutrons and on the inventory charges. From prior work, it appeared that a length of 20 ft and a void fraction of 25% are nearly optimum (43). Accordingly, these values were adopted.

It turns out that the temperatures and temperature differences in the reactors studied are rather far from the limiting values, or can easily be controlled by redistributing graphite among fuel plates, fuel element case, and permanent structure. The end-leakage of neutrons (2%), though appreciable, is not large, and if the end-blankets are not less than 90% efficient, the losses will be no greater than 0.004 neutrons per neutron absorbed in fissile atoms. Thus, there is little incentive to lengthen the reactor, either to save neutrons or to reduce the temperatures and temperature differences. On the other hand, as shown in Appendix A, the power removal turns out to be independent of the length when the fuel plates are operated at maximum allowable thermal stress, and hence there is no strong incentive toward a shorter core either.

Length of core and volume fraction of gas passages having been assigned, concentrations of moderator and fuel in the core were chosen such that the moderator-fuel ratios, $N(C)/N(U)$, lay in the range from 4,000 to 10,000 as recommended by Perry (43) and Chernick (84). Next, the critical diameters were found by means of one-dimensional, 34-group calculations with the GNU reactor program for the IBM-704. These calculations were performed with fission products and higher isotopes present in either estimated equilibrium amounts or in token amounts. The result of the calculation was a first estimate of the critical diameter together with data from which the reaction rate coefficients of the various isotopes could be computed (Sec. 3.2.3).

The operating pressure was computed from the diameter, thickness, and working stress specified for the reactor vessel. The power capability was then computed by means of Eq. A-11 and A-24, Appendix A, which relates the power in Mwt to the key specifications listed below.

Key Parameters for Thermal Design of Gas-Cooled
Graphite-Moderated Breeder Reactor

Symbol	Item	Value
L	Length of core, ft	20
T_1	Inlet temperature of helium, $^{\circ}\text{R}$	985
T_2	Outlet temperature of helium, $^{\circ}\text{R}$	1960
R_1	Ratio of pump-work to reactor heat	0.027
R_2	Ratio of pump-work expended in core to total pump-work	0.75
F_1	Gas-passage volume fraction in core	0.25
F_2	Fuel-plate volume fraction in core	0.16
t_2	Thickness of fuel plates, in.	0.185
d	Diameter of core, ft	4.6
D	Diameter of pressure vessel, ft	12.6
P	Gas pressure, psi	2200
f	Fanning friction factor	0.025
R_3	Ratio of maximum power density in core to radial-average power density	1.8
c_p	Heat capacity of helium, Btu/lb $^{\circ}\text{R}$	1.248

~~In the table, variables through t (the fuel plate thickness) are in-~~
dependent. The variables L, T_1 , T_2 , F_1 have been fixed by prior considerations. It is common practice to allow about 10% of the gross electrical power for internal usage in gas-cooled reactor plants. Allowing 3-1/2% for station auxiliaries leaves 6-1/2% for pumping the coolant, which, reduced by the thermodynamic efficiency of 40% gives ~ 0.027 as the ratio of pump-work to reactor heat.

The choice of R_2 , the ratio of pump work expended in the core to total pump work, should be optimized with respect to the net power cost, with power output balanced against capital cost of the heat exchangers. This optimization lies outside the scope of the present work, and it was necessary to make an arbitrary choice. It was elected to keep the core structure compact and

expend most of the pump-work there, and to design the heat exchangers with somewhat larger frontal areas than would otherwise be necessary.

The remaining independent variables must be chosen in such a way that the stress in the fuel plates shall not exceed the limiting value. The maximum allowable temperature is not known, but the available evidence indicates that temperatures up to 4000°F may be tolerable (126), if UO_2 is stable in graphite at that temperature; if not, the fuel could be reduced to UC during the fabrication process. The thermal stress limit is thought to be about 1000 psi. This is not a restrictive limit unless the thermal conductivity of graphite falls as a result of radiation damage more than was estimated (Appendix A).

These same variables must also be chosen to give an integral number of reactors per station, and a practical set of dimensions for the fuel element. Fortunately, the mechanical and thermal properties of graphite are such that " F_2 " and " t_2 " may be varied over a considerable range - that is, the distribution of graphite between permanent structure, fuel-element cases, and fuel-plates is not restricted to a narrow range. For the specifications cited in Table 6.4.1, a value of 0.16 for F_2 and 0.185 for t_2 results in a core producing 603 Mwt and a blanket producing 89 Mwt. Four reactors can produce 1000 Mwe when the overall plant efficiency is 36.1%. When the fuel element has seven fuel plates, these are 3.4 in. wide, and the corresponding lattice spacing is 5.3 in. If the walls of the cases are 0.45 in. thick, the graphite planes forming the permanent lattice structure are 1.0 in. thick. The outside dimensions of the fuel element are 4.3 in. by 4.3 in., and whatever length is convenient and desirable from the standpoint of handling, say, about 24 in.

The power, together with selected values of processing rates for fuel and fertile streams, holdup times, and reaction rate coefficients, was used in a calculation of the equilibrium reactor by means of the ERC-5 program (Appendix H). When the equilibrium concentrations were much different from those used in computing the initial values used in the GNU calculation, the sequence of steps was repeated. This was seldom necessary, however, as the reaction rate coefficients are insensitive to minor changes in the

Table 6.4.1. GGBR Nuclear Characteristics
Station Power: 1000 Mwe Fuel Processing: Continuous, on-site

	Case Number							
	1	2	3	4	5	6	7	8
	Key Variables							
Moderator-fuel ratio, N(C)/N(U)	7700	7900	7800	8100	7900	7700	8200	8400
Thorium inventory, tonnes	100	100	100	100	100	100	150	200
Fuel-stream processing cycle time, days	37.0	27.7	18.5	10.0	18.5	18.5	18.5	18.5
Fertile-stream processing cycle time, days	217	217	217	217	362	145	326	444
	Nuclear Economy							
Net neutron yield, $\bar{\eta}_e$	2.2266	2.2282	2.2290	2.2310	2.2278	(2.2303) ^b	2.2311	2.2316
Nuclear losses ^a :								
U-233 absorptions	0.9001	0.9029	0.9049	0.9105	0.9031	0.9084	0.9109	0.9128
U-235 absorptions	0.0999	0.0971	0.0951	0.0895	0.0969	0.0916	0.0891	0.0871
Th-232 fissions	0.0064	0.0061	0.0059	0.0063	0.0074	0.0048	0.0063	0.0064
Pa-232 absorptions (x 2)	0.0370	0.0381	0.0394	0.0404	0.0411	0.0378	0.0290	0.0224
U-236 absorptions	0.0160	0.0144	0.0127	0.0108	0.0141	0.0114	0.0118	0.0118
Np-237 absorptions	0.0024	0.0017	0.0012	0.0005	0.0011	0.0009	0.0010	0.0010
Xenon absorptions	0.0500	0.0496	0.0461	0.0462	0.0436	0.0472	0.0454	0.0447
Samarium absorptions	0.0089	0.0088	0.0086	0.0084	0.0086	0.0086	0.0087	0.0087
Other fission product absorptions	0.0388	0.0298	0.0186	0.0121	0.0206	0.0210	0.0209	0.0210
Core moderator absorptions } Fuel carrier absorptions }	0.0389	0.0389	(0.0389) ^b	0.0390	(0.0389) ^b	(0.0389) ^b	0.0390	0.0390
Thorium carrier absorptions	-	-	-	-	-	-	-	-
Corrosion products	-	-	-	-	-	-	-	-
Delayed neutrons	-	-	-	-	-	-	-	-
Leakage	0.0046	0.0046	0.0033	0.0046	0.0049	0.0024	0.0046	0.0046
Allowance for additional end-losses	0.0040	0.0040	0.0040	0.0040	0.0040	0.0040	0.0040	0.0040
Processing losses	0.0035	0.0043	0.0057	0.0095	0.0063	0.0057	0.0058	0.0058
Net breeding gain	0.0161	0.0279	0.0446	0.0492	0.0372	0.0476	0.0546	0.0623
	Other Characteristics							
$\xi \Sigma_s / N(U)$, barns/atom (U-233+U-235)	6500	6500	6500	6500	6500	6500	6500	6500
Per cent fissions in fertile stream	10.0	10.3	10.0	10.8	16.8	6.8	12.0	12.8
Per cent thermal fissions	77	77	77	77	77	77	77	77

- a. Neutrons or fissile atoms lost per neutron absorbed in fissile atom
b. Error in calculations adjusted by comparison with neighboring cases.

spectrum of neutrons. The ERC-5 program computes inventories of various isotopes of interest, production rates, and a neutron balance for the equilibrium reactor. The fuel yield and the fuel cycle cost are readily calculated therefrom.

By means of preliminary calculations not reported here, it was found that concentrations of moderator and fuel giving $N(C)/N(U)$ ratios in the range 7000 to 8000 resulted in a broad maximum in the difference between the net neutron yield and absorptions in moderator. The corresponding diameter of the critical core was about 4.6 ft, and of the reactor vessel about 13 ft. This size, in combination with a wall thickness limitation of 10 in. fixed the operating pressure at 2200 psi.

Other variables which were considered in various calculations were the thorium inventory, fertile stream processing cycle time, and the fuel stream processing cycle time (or burnup).

6.4.3 GGBR Results

Detailed results are given for eight cases in Table 6.4.1, etc. Cases 1 through 4 constitute a series in which the thorium inventory was 100 tonnes, the fertile stream processing cycle time was 217 days, and the fuel stream processing cycle time was varied from 37 down to 10 days. The pellets discharged from the blanket contained about 6 grams of U-233 and Pa-233 per kg of thorium. The burnup (fraction of U-233 in fuel plates that is destroyed) varied from 0.56 down to 0.21.

As the burnup decreased, net breeding gain increased from 0.016 to 0.049. This resulted mainly from a decrease in neutron losses to fission products, especially those other than to xenon and samarium (which are near saturation and not much affected by the residence time). Losses to Pa-233 increase, partly because more of it is formed, and partly because the leakage of neutrons to the blanket increases as the concentration of fission products in the core falls. In spite of this (and the resulting greater rate of formation of U-234), the proportion of fissions in U-235 decreases. This comes about because as the breeding gain rises, the rate of removal of U-235 via sales increases, and its equilibrium concentration falls. Losses to U-236 also decrease for the same reason.

The inventory of fissile isotopes increases as the fuel stream cycle time decreases (Cases 1 - 4, Table 6.4.2). The increase is almost entirely in the fuel stream processing plant and in the fuel reserve - both of these being proportional to the processing rate. The station inventory more than doubles, rising from about 900 to over 1900 kg. As the fuel stream processing cycle time is decreased, the rate of increase of breeding gain, at first, is greater than that of inventory, and the fuel yield rises. However, at a burnup somewhat less than 35%, the inventory begins to rise more rapidly while the increase in breeding gain becomes marginal. The fuel yield passes through a maximum of about 3-1/4% per year (Table 6.4.3).

The processing costs increase monotonically as the fuel-stream processing cycle time falls, and the charges on the inventory of fuel isotopes also increase. As indicated in Table 6.4.3, the cost rises from 1.3 to 1.6 mills/kwhr.

In a second series of calculations (Cases 5, 3, and 6), the thorium inventory was 100 tonnes, the fuel stream processing cycle time was 18.5 days (burnup about 35%), and the fertile stream processing cycle time was varied from 362 down to 145 days. The loading of the product from the blanket fell from 9.8 to 4.4 grams of uranium plus protactinium per kg of thorium. The breeding gain was not much affected, rising from 0.037 to 0.048 (Table 6.4.1). There were appreciable changes in the fissile inventories, however. Inventory of U-235 in the blanket fell from nearly 400 kg at the longer cycle times to about 100 kg. The holdup in the fertile stream processing rose from 65 to 126 kg. There was a decline in total inventory of about 200 kg. This, coupled with a rising breeding gain, caused the fuel yield to increase from 2.3 to 3.4 (Table 6.4.3). The fuel cycle costs also increased from 1.3 to 1.5 mills/kwhr, mainly because of increased cost of the fertile stream processing. It is clear that efforts to increase the fuel yield further by increasing the rate of fertile stream processing will result only in marginal gains and that the costs will rise steeply.

In a third series of calculations (Cases 3, 7, and 8), the fuel stream cycle time was 18.5 days, the thorium inventory was varied from 100 to 200 tonnes, and the fertile stream processing cycle time was varied to maintain a constant loading of about 6 grams of uranium and protactinium per kg of thorium

Table 6.4.2. GGBR Inventories and Processing Rates

	Case Number							
	1	2	3	4	5	6	7	8
Key Variables								
Station thorium inventory, tonnes	100	100	100	100	100	100	150	200
Fuel stream processing cycle time, days	37.0	27.7	18.5	10.0	18.5	18.5	18.5	18.5
Fertile stream processing cycle time, days	217	217	217	217	362	145	326	444
Station Inventories								
Station fissile inventory, kg								
U-233 in fuel stream	116	112	112	112	112	116	108	108
U-235 in fuel stream	16	16	16	12	16	12	12	12
U-233 in fuel-stream processing	188	280	460	848	468	460	448	440
U-235 in fuel-stream processing	40	52	72	128	84	68	72	68
U-233 in fuel-stream reserve	136	164	224	372	216	232	220	216
U-235 in fuel-stream reserve	12	16	24	40	28	20	24	24
U-233 in fertile stream	208	212	216	216	392	124	368	536
Pa-233 in fertile stream	96	96	100	100	108	90	108	112
U-233 in fertile-stream processing	91	91	92	94	65	124	68	84
Pa-233 in fertile-stream processing	24	25	18	24	15	34	17	14
Total	927	1064	1334	1946	1504	1280	1445	1614
Processing Rates								
Fresh fuel fabrication rate ^a , kg/day	5.0	6.1	8.5	13.7	8.5	8.8	7.9	7.8
Excess fissile uranium produced, kg/day	0.041	0.072	0.123	0.126	0.110	0.126	0.140	0.160
Thorium processing rate, kg/day	443	443	442	443	288	660	441	438
Thorium replacement rate, kg/day	11.5	11.5	11.5	11.5	7.95	16.2	7.67	5.75
Graphite replacement rate, kg/day	298	398	596	1100	596	596	596	596
Fertile stream loading, g(U+Pa)/kg Th	6.2	6.2	6.3	6.4	9.8	4.4	6.3	6.3

a. Includes only U-233 and U-235 in fresh fuel elements.

Table 6.4.3. GGBR Fuel Costs and Performance Summary

	Case Number							
	1	2	3	4	5	6	7	8
	Key Variables							
Station thorium inventory, tonnes	100	100	100	100	100	100	150	200
Fuel-stream processing cycle time, days	37	27.7	18.5	10	18.5	18.5	18.5	18.5
Fertile-stream processing cycle time, days	217	217	217	217	362	145	326	444
	Cost Breakdown, mills/kwhr							
Uranium inventory	0.079	0.091	0.115	0.116	0.129	0.112	0.123	0.138
Thorium inventory	0.049	0.049	0.049	0.049	0.049	0.049	0.074	0.098
Graphite inventory	0.001	0.001	0.001	0.001	0.001	0.001	0.001	0.001
Fuel stream processing	0.326	0.368	0.450	0.595	0.451	0.459	0.434	0.430
Fertile stream processing	0.752	0.752	0.752	0.752	0.668	0.855	0.752	0.753
Thorium replacement	0.016	0.016	0.016	0.016	0.011	0.023	0.016	0.016
Graphite replacement	0.055	0.073	0.109	0.202	0.109	0.109	0.109	0.109
Breeding credit ^b	0.032	0.056	0.089	0.098	0.074	0.093	0.109	0.125
	Performance Summary							
Net fuel cycle cost ^b , mills/kwhr	1.25	1.29	1.40	1.68	1.34	1.52	1.40	1.42
Fuel yield ^b , %/yr	1.92	2.50	3.22	2.39	2.32	3.39	3.58	3.70
Fuel specific power, Mwe/kg	1.1	0.94	0.75	0.51	0.67	0.78	0.69	0.62
Thorium specific power, Mwe/tonne	10	10	10	10	10	10	7	5
Net breeding gain	0.016	0.028	0.048	0.049	0.043	0.049	0.055	0.062
Per cent of heat from fertile stream	20	20	20	20	27	17	22	23
Per cent burnup of fuel	57	47	34	21	32	35	35	35

a. For cost bases see Sec. 3.2

b. Plant factor: 80%

in the product from the blanket. The principal effect is a decline in losses to protactinium as it is diluted by thorium, and an increase in U-233 inventory in the blanket. In the range investigated, the increase in breeding gain over-balances the inventory increase, and the fuel yields increase. Since the rate of processing the blanket in kg/day varied little, the fuel stream processing declined only slightly (as a result of more fissions in the blanket). The increased charges on the U-233 inventory in the blanket therefore dominated, and the net cost increased in spite of increased breeding credit (Table 6.4.3).

Selected cases are plotted in Fig. 6.4.1; the small numbers near the points are the corresponding case numbers. The curve through points 3, 7, and 8 show the variation of fuel yield and fuel cost with increasing thorium inventory, as discussed above. Similar curves through the other points, while not supported by calculation, are plausible predictions of the behavior. Comparison of cases 3 and 5 in Table 6.4.3 suggests that about half a mill could be saved in case 9 by increasing the fertile stream cycle time to 360 days (loading of about 10 grams of fuel isotopes per kg of thorium). Taking the above into account, the curve labeled "estimated locus of maximum performance" was drawn. It was concluded that the GGBR as presently conceived, is capable of fuel yields up to 4% per year at fuel cycle costs not greater than 1.5 mills/kwhr, and costs as low as 1.2 mills/kwhr can be obtained with positive breeding gain (possible as high as 2% per year).

Further improvement in the performance of the GGBR would depend on changes in the conceptual design and design assumptions. The most promising avenue is the development of a fuel-matrix from which the volatile fission products can escape quickly into the gas stream whence they may be removed by adsorption on charcoal. This might add as much as 0.04 to the breeding gain and would put the performance of the GGBR on a par with the MSBR.

Substantial benefits would accrue from the development of alternate processes for purifying the fuel and fertile streams. The use of Thorex to process the fuel stream requires a 90-day holdup of spent fuel before it is sent to the extraction plant. This results in a large inventory of fuel, especially at the higher processing rates (low burn-ups). An alternate process, such as the leaching of the ground-up fuel plates with molten salt,

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followed by fluorination to recover UF_6 , would, if feasible, obviate the necessity of the 90-day holdup. Further, if ThF_4 pellets could be used in the blanket instead of ThO_2 , these might be processed rapidly by fluoride volatility process, thus reducing the inventory of U-233 in the blanket and decreasing losses to Pa-233 (through reduced fissions in the blanket). These processes might well be cheaper than Thorex, and so lower the fuel cost as well as increase the fuel yield.

6.5 Deuterium-Moderated Gas-Cooled Breeder Reactor (DGBR)

6.5.1 DGBR Special Assumptions

Although the DGBR core is strictly heterogeneous from a nuclear standpoint, it was feasible to perform only a few calculations on a heterogeneous model. These, however, made possible an extrapolation of the calculations based on a homogeneous model, in which the materials in the core were "homogenized".

The calculations based on a heterogeneous model were similar to those performed for the MSBR and described in Sec. 6.2 (see also Appendix E). Disadvantage factors were computed for a multi-region unit cell by means of the Cornpone program (91) for the ORACLE. These were used in a one-dimensional multi-group calculation of the finite cylindrical reactor consisting of core, blanket, and reflector, to determine critical concentrations and reaction rate coefficients for the various elements in the nuclear active zones. The axial leakage from the bare-ended reactor was distributed among absorptions in blanket isotopes, etc., in the same proportions that radial leakage was ~~computed to be distributed by the code. The equilibrium calculations were~~ performed using the ERC-5 program in the same manner as those for the GGBR.

The analysis of the heat transfer in the core was performed using the ~~working equations developed in Appendix A, appropriately modified for the~~ cylindrical geometry of the fuel elements. The ratio of blower electrical input to core thermal rating was taken as 0.025, and the ratio of channel pressure drop to total circuit pressure drop was 0.75.

6.5.2 DGBR Key Variables

The optimization of the core of the DGBR is appreciably more complex and difficult than that of any of the other concepts studied. Degrees of freedom in the thermal design are much more restricted than in the GGBR for the reason that the volume and surface area of the available fuel carrier are much smaller. The volume fractions of moderator, Zircaloy, fuel carrier, and coolant must be adjusted simultaneously to provide adequate heat transfer surface

coolant flow area, favorable moderator-fuel atom ratios, and tolerable parasitic captures in zirconium, moderator, and fuel carrier. These conflicting requirements were resolved as far as possible in a preliminary series of calculations not reported in detail; an exemplary set of compatible specifications is given in Table 6.5.1.

Parasitic captures in Zircaloy are controlled by maintaining a low Zr-fuel atom ratio. Hence, it is advantageous to lump the fuel into a few channels. The fuel elements tend to become large, however, because the coolant flow area per element must increase in proportion to the fuel content if the specific power is to be preserved. Although not studied, improvement with respect to parasitic captures in Zircaloy is limited in a practical sense by size and weight of fuel elements that can be handled conveniently. Further, as the blackness of the fuel with respect to neutrons increases, the critical mass tends to increase with adverse effects on fuel yield and fuel cycle cost, although there is some compensatory reduction in parasitic captures in fission products and improvement in neutron yield.

The average neutron yield, $\bar{\eta}\epsilon$, increases monotonically, but at a diminishing rate, as the moderator-fuel ratio increases. Parasitic captures in the moderator (D_2O) containing 0.25% H_2O also increase. The moderator-fuel ratio may be varied either by varying the lattice spacing of the fuel channels or by varying the concentration of fissile isotopes in the fuel carrier. The concentration can not be made arbitrarily small, however, because the residence times required to achieve burnups in the range 40 to 50% become impractically small. A lower concentration of fuel is fixed, moreover, by the necessity of maintaining a low zirconium-fuel ratio as discussed above. Thus, the moderating ratio was increased by increasing the lattice spacing.

If the diameter of the core is already at a limit fixed by pressure vessel design considerations, then increasing the lattice pitch is accompanied by a decrease in the number of channels, and a corresponding decrease in power capability. If carried too far, it becomes impractical to compensate this effect by other changes, such as increases in gas pressure, pump-work, etc., and it becomes necessary to increase the number of reactors in a station. This not only increases the capital costs, but increases the fuel cycle costs through increased use charges on uranium, thorium, and heavy water.

Table 6.5.1. DGBR Nuclear Characteristics
 Station Power: 1000 Mwe Fuel Processing: Continuous, on-site

	Case Number							
	1	2	3	4	5	6	7	8
	Key Variables							
Moderator-fuel ratio, $N(D)/N(U)$	3000	3000	3000	3000	3800	3800	3000	3200
Thorium inventory, tonnes	200	130	175	150	130	130	130	130
Fuel-stream processing cycle time, days	112	112	112	108	82	112	153	150
Fertile-stream processing cycle time, days	280	180	393	414	504	504	292	274
	Nuclear Economy							
Net neutron yield, $\bar{\eta}$	2.2276	2.2273	2.2264	2.2257	2.2301	2.2296	2.2256	2.2325
Nuclear losses ^a to:								
U-233 absorptions	0.9232	0.9222	0.9194	0.9165	0.9134	0.9116	0.9166	0.9193
U-235 absorptions	0.0768	0.0778	0.0806	0.0835	0.0866	0.0884	0.0834	0.0807
Th-232 fissions	0.0012	0.0012	0.0012	0.0012	0.0013	0.0013	0.0012	0.0013
Pa-233 absorptions	0.0171	0.0236	0.0201	0.0242	0.0289	0.0284	0.0253	0.0260
U-236 absorptions	0.0073	0.0071	0.0083	0.0089	0.0094	0.0103	0.0088	0.0080
Np-237 absorptions	0.0007	0.0006	0.0007	0.0008	0.0008	0.0012	0.0010	0.0009
Xenon absorptions	0.0480	0.0485	0.0474	0.0475	0.0482	0.0482	0.0479	0.0483
Samarium absorptions	0.0087	0.0087	0.0087	0.0087	0.0088	0.0088	0.0089	0.0088
Other fission product absorptions	0.0226	0.0227	0.0225	0.0226	0.0236	0.0302	0.0301	0.0299
Moderator absorptions ^b	0.0270	0.0270	0.0270	0.0270	0.0345	0.0345	0.0270	0.0300
Fuel carrier	0.0012	0.0012	0.0012	0.0012	0.0015	0.0015	0.0012	0.0012
Thorium carrier ^c	0.0008	0.0008	0.0008	0.0008	0.0010	0.0010	0.0008	0.0010
Corrosion products								
Delayed neutrons								
Leakage	0.0014	0.0014	0.0014	0.0018	0.0028	0.0028	0.0014	0.0031
Allowance for additional end-losses	0.0040	0.0040	0.0040	0.0040	0.0040	0.0040	0.0040	0.0040
Processing losses	0.0056	0.0056	0.0056	0.0056	0.0056	0.0045	0.0045	0.0045
Net breeding gain	0.0820	0.0749	0.0775	0.0714	0.0597	0.0529	0.0635	0.0655
$\xi \Sigma_s / N(U)$, barns per atom (U-233+U-235)	8200	8200	8200	8300	10,500	10,500	8200	8700
Per cent fissions in fertile stream	6.7	5.7	11.0	13.6	20.1	19.8	10.0	10.2
Per cent thermal fissions	84.2	84.2	84.2	84.2	86.5	86.5	84.2	85.1

- a. Neutrons or fissile atoms lost per neutron absorbed in fissile atom
 b. Includes absorptions in D_2O , H_2O , and Zr in core
 c. Includes absorptions in D_2O , H_2O , and Zr in blanket

By means of preliminary calculations, it was found that the difference between the neutron yield and moderator captures passes through a broad maximum at slowing down powers, $\xi \Sigma_s / N(U)$, in the range from 8,000 to 12,000. Further trials to find a compatible set of specifications that would give good nuclear characteristics and promising thermal performance were performed. The set of specifications given below appeared to satisfy these criteria, and are believed to be not far from the optimum conditions.

Key Parameters for Thermal Design of
Deuterium-Moderated Gas-Cooled Breeder Reactor

Symbol	Item	Value
L	Length of core, ft	20
$\mathcal{L}$	Lattice pitch, in.	14
T_1	Inlet temperature of helium, $^{\circ}\text{R}$	988
T_2	Outlet temperature of helium, $^{\circ}\text{R}$	1963
R_1	Ratio of pump work to reactor heat	0.025
R_2	Ratio of pump work expended in core to total pump work	0.25
F_1	Gas-passage volume fraction in core	0.0917
F_2	Fuel tube volume fraction in core	0.0193
t	Thickness of fuel tubes, in.	0.125
D	Diameter of core, ft	7.27
P	Gas pressure, psi	2000
F	Fanning friction factor	0.025
R_3	Ratio of maximum power density in core to radial-average power density	1.75
C_p	Heat capacity of helium, Btu/lb $^{\circ}\text{R}$	1.248

6.5.3 DGBR Results

The important independent variables to be optimized with respect to fuel yield and fuel cycle cost were the fuel and fertile stream processing cycle times, and the thorium inventory. Several sets of calculations in which these were varied independently were performed and are reported in detail in reference 78. Detailed results for illustrative cases are listed in Table 6.5.1, etc. The largest breeding gain calculated was 0.082 (Case 1). Here the thorium inventory was 200 tonnes, the fuel stream cycle time was 112 days, and the fertile stream cycle time was 280 days. The largest single loss was to xenon (~ 0.05 neutrons per neutron absorbed in fissile atom). This loss is near the saturation value, and is not a function of the fuel stream cycle time, unless this were to become impractically short. The best hope for reducing it lies in developing and demonstrating fuel matrices which will permit the rapid diffusion of xenon, and its precursor, iodine, into the coolant gas.

The loss in breeding gain to the moderator amounts to 0.027. This could be reduced by decreasing the moderator-fuel atom ratio but the gain would be more than offset by decreases in the neutron yield, $\bar{\nu}$, from its value of 2.228. Losses to "other fission products" (0.023), can be reduced by shortening the fuel-stream cycle time, but the concomitant increase in inventory of fissile atoms in fuel processing results in a decrease in the fuel yield. This aspect of the optimization is discussed more fully below.

Losses to protactinium (0.017) are modest, and as shown below, further savings brought about by increasing the thorium inventory are offset by increased inventory of fissile isotopes in the blanket. It is concluded that a breeding gain of 0.08 in the DGBR is about the best that can be achieved if xenon cannot be removed from the fuel.

The dependency of fuel yield and fuel cycle cost on thorium inventory in the range from 130 to 200 tonnes was studied at fertile stream loadings ranging from 4 to 11 g of U-233 + Pa-233 per kg thorium. The fuel stream cycle time was adjusted to maintain a burnup of about 40%. The trends in breeding gain and inventory are similar to those of the reactor concepts reported in previous sections. The breeding gain increases at a diminishing

rate as the inventory of thorium increases, and the inventory of fissile isotopes increase steadily. As a result, the fuel yield passes through a maximum; at the higher fertile stream loadings, the maximum occurs between 150 and 175 tonnes of thorium, at the lower loadings, it shifts toward 200 tonnes. The cost increases steadily, of course, mainly because of increased charges on thorium and uranium. The increase amounts to about 1/3 mill per kwhr between 130 and 200 tonnes of thorium, and the variation in fuel yield in this range is only 0.1 to 0.2 per cent per year. The behavior is well illustrated by the small loops in Fig. 6.5.1, one of which is indicated by the arrow labeled "A". The small numbers near some of the points correspond to case numbers in Table 6.5.1, etc.

The fuel yield and the fuel cycle cost vary more strongly with the fertile stream loading than with the thorium inventory. This is clearly shown in Fig. 6.5.1 where the loading increases in the direction indicated by the arrow labeled "B" from 4.2 to 10.8 g (U+Pa) per kg Th. The fuel yield falls from 4.6 to about 3.6 per cent per year, partly because the losses to protactinium in the blanket increase (compare cases 1 through 4) and losses to moderator increase mainly because the inventory of fissile isotopes in the blanket increases (Table 6.5.2). The increased cost associated with increased inventories of thorium and fissile isotopes is more than compensated by the decrease in fertile stream processing cost at the higher loadings, and the fuel cycle cost falls from about 1.6 to 1.3 mills per kwhr as the blanket fissile loading increases.

The dashed curves in Figure 6.5.1 show the dependency of fuel yield and cost on the fuel stream processing cycle time. In the cases correlated by the curve at the left, and indicated by the arrow labeled "C", the thorium inventory was 130 tonnes, the fertile stream cycle time was 500 days, and the fuel stream cycle time was varied from 35 to 150 days. The corresponding burnups ranged from 20 to 60%. It is seen that the optimum burnup is around 50% (assuming this can be achieved in practice). Higher burnups, because of higher inventories in fuel-stream processing, result in lower fuel yields. The loss in breeding credit largely offsets savings in fuel-stream processing costs, and cost decreases are marginal (Table 6.5.3). At burnups below 40%, savings

Table 6.5.2. DGBR Inventories and Processing Rates

	Case Number							
	1	2	3	4	5	6	7	8
	Key Variables							
Station thorium inventory, tonnes	200	130	175	150	130	130	130	130
Fuel stream processing cycle time, days	112	112	112	108	82.	112	153	150
Fertile stream processing cycle time, days	280	180	393	414	504	504	292	274
Station-D ₂ O inventory, tonnes	165	160	164	162	160	160	160	160
	Station Inventories							
Station fissile inventory, kg								
U-233 in fuel stream	536	541	506	489	360	361	511	522
U-235 in fuel stream	49	49	51	54	44	45	53	52
U-233 in fuel-stream processing	425	424	422	422	419	277	280	281
U-235 in fuel-stream processing	51	51	55	59	66	49	43	40
U-233 in fuel-stream reserve	190	191	178	178	170	170	179	186
U-235 in fuel-stream reserve	17	18	18	19	21	21	18	19
U-233 in fertile stream	327	182	477	511	635	630	327	313
Pa-233 in fertile stream	110	100	115	119	127	126	108	112
U-233 in fertile-stream processing	91	119	76	74	68	68	88	92
Pa-233 in fertile-stream processing	28	35	30	28	31	30	32	28
TOTAL	1824	1710	1928	1953	1941	1777	1639	1645
	Processing Rates							
Fresh fuel fabrication rate ^a , kg/day	8.00	7.98	7.98	8.03	8.01	6.35	6.35	6.34
Excess fissile uranium produced, kg/day	0.228	0.205	0.266	0.215	0.194	0.171	0.183	0.189
Thorium processing rate, kg/day	781	769	503	407	297	296	498	534
Thorium replacement rate, kg/day	18.2	17.9	12.9	11.0	9.0	9.0	12.7	13.5
Graphite replacement rate, kg/day	26.1	25.8	27.4	28.2	43.1	27.5	19.9	19.5
Fertile stream loading, g(U+Pa)/kg Th	4.25	4.21	6.46	7.91	10.8	10.7	6.40	6.24

a. Includes only U-233 and U-235 in fresh fuel elements.

Table 6.5.3. DGBR Fuel Costs and Performance Summary

	Case Number							
	1	2	3	4	5	6	7	8
Key Variables								
Station thorium inventory, tonnes	200	130	175	150	130	130	130	130
Fuel-stream processing cycle time, days	112	112	112	108	82.	112	153	150
Fertile-stream processing cycle time, days	280	180	393	414	504	504	292	274
Cost Breakdown ^a , mills/kwhr								
Uranium inventory	0.156	0.147	0.165	0.167	0.166	0.152	0.141	0.141
Thorium inventory	0.098	0.064	0.086	0.073	0.064	0.064	0.064	0.064
Deuterium inventory	0.185	0.179	0.183	0.181	0.179	0.181	0.179	0.179
Fuel stream processing	0.432	0.432	0.432	0.432	0.432	0.376	0.377	0.380
Fertile stream processing	0.905	0.905	0.788	0.734	0.672	0.672	0.788	0.796
Thorium replacement	0.026	0.025	0.018	0.015	0.013	0.013	0.018	0.019
Graphite replacement	0.006	0.006	0.006	0.006	0.010	0.006	0.005	0.004
Deuterium replacement	0.009	0.009	0.009	0.009	0.009	0.008	0.008	0.008
Breeding credit ^b	0.179	0.161	0.177	0.168	0.151	0.134	0.143	0.148
Performance Summary								
Net fuel cycle cost ^b , mills/kwhr	1.64	1.61	1.51	1.45	1.39	1.34	1.44	1.44
Fuel yield ^b , %/yr	4.60	4.42	4.31	4.04	3.67	3.54	4.11	4.21
Fuel specific power, Mwe/kg	0.55	0.59	0.52	0.51	0.51	0.56	0.61	0.61
Thorium specific power, Mwe/tonne	5.0	7.7	5.7	6.7	7.7	7.7	7.7	7.7
Net breeding gain (rounded)	0.082	0.075	0.078	0.071	0.060	0.053	0.064	0.066
Per cent of heat from fertile stream	18	17	22	22	30	30	21	21
Per cent burnup of fuel	40	40	40	40	40	50	50	50

a. For cost bases see Sec. 3.2

b. Plant factor: 80%

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associated with lower fission product poisoning become marginal, while the inventory of fissile isotopes in the fuel stream processing increases rapidly. This drives the fuel yield down. Meanwhile, increased charges on the fissile inventory and increased fuel stream processing costs drive the fuel cycle cost up rapidly. The other dashed curves in Figure 6.5.1 are similar except the calculations were performed with fuel stream cycle times resulting in burnups of 40 and 50%. It is seen that these, taken together with the solid curves, define rather well a locus of maximum performance for the DGBR.

The calculations reported thusfar were performed using a "homogenized" nuclear model for the core -- the heterogeneity was neglected. Two cases were selected for detailed study taking the heterogeneity into account. These two cases are represented by two solid points in Figure 6.5.1, and their relation to the "homogenized" model are indicated by arrows joining the corresponding points. There is a slight improvement in performance. The "heterogeneous" model generates a slightly more thermal spectrum of neutrons, with a consequent slight improvement in the neutron yield, $\bar{\eta}_e$ (compare cases 7 and 8). But the fuel is also less effective, and the predicted critical mass is greater. This increases the cost. The net effect is to shift the locus of maximum performance slightly to the right. Taking this into account, and making some allowance for possible further optimization of moderator-fuel ratio, cycle times, and thorium inventory, the curve labeled "estimated locus of maximum performance" was drawn. On the basis of this curve, it is predicted that the DGBR is capable of achieving fuel yields up to 4.5%/yr at a fuel cycle cost not greater than 1.5 mills/kwhr, and that costs as low as 1.25 mills/kwhr at fuel yields of 3% per year can be realized. The curve is still trending downward in this range, but it is doubtful that lower costs can be achieved because the fraction of fissions in the fertile stream is rising, and the fraction of heat that must be removed from the blanket moderator at low temperature becomes impractically large.

7. COMPARISON OF REACTOR CONCEPTS

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7. COMPARISON OF REACTOR CONCEPTS

The various concepts studied may be compared on three different bases: (a) nuclear capability as measured by the maximum fuel yield, (b) economic potential as measured by the minimum fuel cycle cost at prevailing prices, and (c) relative status of development work. The nuclear capability is indicative of the competitive strength of a reactor concept in the long run when other sources of energy become inadequate or relatively more costly, and nuclear fuels consequently become more valuable. The economic potential is a criterion in the selection of a concept for intensive development now, as is also the relative status of present development.

7.1 Nuclear Capability

In all the concepts studied, the fuel cycle cost reached a maximum, or increased very rapidly with increasing fuel yield, in the range of fuel cycle costs from 1.5 to 2 mills/kwhr. This range of costs lies below those quoted for current plants of conventional design (i.e., clad fuels, low conversions), which range down to about 2 mills/kwhr, and also below that predicted for the natural-uranium-fueled reactor proposed by W. B. Lewis (133), for which fuel cycle costs down to 2.2 mills/kwhr have been predicted, based on U. S. pricing procedures (138).

7.1.1 Comparison of Systems

For the sake of clarity and uniformity, the fuel yield for each concept at a fuel cycle cost of 1.5 mills/kwhr was selected as a measure of nuclear capability, as shown in Table 7.1.1, together with corresponding values of the breeding gain and the specific powers. For every case studied, the fuel cost was rising steeply in the region of 1.5 mills/kwhr, and the fuel yields listed may be regarded as the maximum that can be obtained at fuel costs of interest.

It is clear that the AHBR exhibits a nuclear capability appreciably superior to the other concepts at reasonable and attractive fuel cycle costs. This superiority arises not only from the magnitude of the predicted fuel yield, but also from the confidence with which high performance breeding

Table 7.1.1 Maximum Fuel Yield for Five Thermal
Breeder Reactors at Fuel
Cycle Cost of 1.5 mills/kwhr

Concept:	AHBR	MSBR	LBRR	GGBR	DGBR
Thorex Processing	Yes	No	Yes	Yes	Yes
Specific Power, Mw/kg					
Thermal	4.0	2.7	1.2	2.7	1.2
Electrical	1.1	1.1	0.5	1.1	0.6
<u>Calculated Breeding Gain</u>					
Neutron Yield, $\bar{\eta}$	2.254	2.214	2.208	2.232	2.228
Nuclear Losses					
Moderator	0.034	0.029	0.025	0.039	0.027
Carriers and Structure	0.029	0.046	0.048	-	0.002
Pa-233 (x 2)	0.031	0.012	0.019	0.023	0.017
Uranium Isotopes	0.008	0.011	0.015	0.013	0.008
Fission Products	0.025	0.027	0.029	0.074	0.079
Processing	0.001	0.002	0.003	0.006	0.006
Miscellaneous	0.015	0.009	0.012	0.015	0.007
Breeding Gain	0.111	0.078	0.057	0.062	0.082
Maximum Fuel Yield, %/yr	16	7	4	4	4.5
Doubling Time @ 80% P. F., yrs	6	14	25	25	22

(i.e., capability of fuel yields of 10% or more and doubling times of 10 years or less) can be predicted. The margin for error and excessive optimism in the basic assumptions is probably of the order of 30%.

The better performance of the AHBR results partly from the superiority of heavy water in a homogeneous system as a moderator (in spite of the necessary Zircaloy structure), and partly from the superiority of fuel processing by means of hydroclones which rapidly and efficiently remove most of the fission products.

The MSBR is the only concept that did not involve the Thorex process at some stage; instead, the fertile stream was treated by means of the fluoride volatility process for rapid and efficient removal of U-233. The superiority of the MSBR to the GGBR and DGBR results from this, and also, in relation to the LBBR, from the superiority of salt compared to bismuth as a heat transfer medium. The reliability of this estimate rests squarely on the validity of the favorable cost assumptions made for the MSBR. If these should prove overoptimistic, and the curve in Fig. 1.1 were shifted up appreciably, the fuel yield at a fuel cycle cost of 1.5 mills/kwhr would decline sharply. The MSBR should be re-evaluated when cost and performance data for the processes postulated become available in quantity and reliability comparable to those associated with the Thorex process.

In the remaining three concepts, showing fuel yields of about 4%/yr, neutron losses to moderator, carriers, structure, or fission products limit the breeding gain, and inventories of fuel isotopes are high in the fertile streams or in fuel stream processing, or both.

For the systems utilizing Thorex, this process was the major factor limiting the fuel yield. Developments which would increase the capacity of the Thorex plant at the same total of fixed and operating charges would permit more rapid processing and a consequent increase in the fuel yields. Although it lay outside the scope of the present study to consider the details of the Thorex process, it is believed that the extrapolation of the costs by Burch, et al. (17) was made with rather less optimism than were the corresponding extrapolations of reactor performance and costs made by this group, and that, relative to other components of fuel cycle costs listed, the Thorex cost is conservatively high.

It should be noted that economies in the Thorex process would not improve the last three concepts studied relative to the AHBR, as that concept would profit as well, but the relative position of the MSBR would suffer. On the other hand, the development of alternate processes, such as the substitution of ThF_4 for ThO_2 as a fertile material, might be feasible in one concept but not in another. But, as concluded in the previous sections, such alternate processes, even when all possible optimism is used, would result in fuel yields not substantially better than those presently predicted for the MSBR, and inferior to those of the AHBR.

7.1.2 Uncertainties in Nuclear Data

The nuclear data for U-233 as of about September 1959 was reviewed and analyzed by Nestor (24). Group average values of eta for U-233 used in this study are given in Appendix F. The corresponding estimates of the nuclear performance are summarized in Fig. 1.1.

Analysis of the data indicates that the value of eta of U-233 is known within $\pm 0.5-0.0\%$ at neutron energies up to 0.8 ev, $\pm 5\%$ from 0.8 to 1.4, and $\pm 10\%$ above 1.4 ev. In order to evaluate the uncertainty in the nuclear performance corresponding to this uncertainty in the nuclear data, the calculations for the AHBR and the MSBR were repeated using values of eta at the upper and lower bounds of the probable error above 0.8 ev. The results are shown in Fig. 7.1.1. It is seen that the uncertainty in the performance of the AHBR is small, but is rather large in the MSBR. This happens because the spectrum of neutrons in the AHBR is more nearly thermal, and the proportion of neutrons captured at epithermal energies in U-233 is less.

The evaluation of the performance of the AHBR is not much affected by the uncertainty. Fuel cycle costs still range down below 1.0 mill/kwhr, and fuel yields up to 15%/yr are attainable.

In the MSBR, the pessimistic assumption for eta of U-233 results in minimum fuel cycle costs of about 1 mill/kwhr, and a maximum fuel yield of only 4% at a cost of 1.5 mills/kwhr.

The effect of this uncertainty on the other three concepts was not computed, but should be similar to that for the MSBR since all of them are

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Fig. 7.1.1 EFFECT OF UNCERTAINTY IN THE
EPITHERMAL VALUE OF ETA FOR U-233 ON
MSBR AND AHBR PERFORMANCE

Plant factor : 0.8
Station power : 1000 Mwe

rather less thermal than the AHBR. The use of the pessimistic assumption would render them incapable of achieving fuel yields of 4%/yr at any fuel cost.

Also, in this study the resonance integral of Pa-233 was assumed to be 1200 barns. Recently published data (139) indicate this assumption was pessimistic and that the integral may be as low as 900 barns. Calculations for the AHBR and the MSBR were repeated using the lower value with the result that in neither system was the performance estimate altered significantly. The maximum fuel yield for the AHBR increased about 0.5%/yr, but the costs were not affected appreciably. In the MSBR, the maximum fuel yield increased by about 0.25%/yr.

7.1.3 Possible Improvements in Fuel Yield

The fuel yield is proportional to the product of the specific power and the breeding gain. The former is inversely proportional to station inventory of fissile isotopes. The fuel yield can be increased by reducing neutron losses and inventory.

AHBR. - An examination of the losses in Table 7.1.1 shows that there ~~is not much hope of effecting further savings of neutrons except possibly~~ by a reduction in leakage of, say, 0.004, and a reduction in losses to protactinium and U-236 of 0.012 (by increasing the thorium inventory to 200 tonnes). The breeding gain might thus be increased to 0.127, but the ~~inventory of fuel would also increase, and it is doubtful that the yield~~ can be increased by more than 10%, if at all, by these means.

A method of processing the fertile stream less expensive than Thorex would increase the capability of the AHBR appreciably by reduction of inventory of U-233 (this increases ^{*}S) and reduction of losses to Pa-233. For example, if pellets of ThF₄, instead of ThO₂, were used, and cooled by helium, with graphite as the blanket moderator and structural material, and if the ThF₄ were processed rapidly by treatment with F₂ to remove isotopes of uranium, fuel yields possibly as high as 20%/yr could be obtained (97).

MSBR. - The only nuclear losses that hopefully might be reduced in the salt reactor are those to protactinium, which might be reduced by further

* S is the specific power

increase in the thorium inventory (and thorium carrier inventory), which would add considerably to the cost. The volume of the external fuel system is already optimistically low, and the inventory of U-233 in the fertile stream could probably not be lowered appreciably except at extremely high and costly processing rates.

Compared to the AHBR, the MSBR suffers from the use of a higher neutron absorbing moderator and coolant (graphite and salt vs D_2O). In the MSBR, nuclear losses to graphite and salt limit the breeding gain (0.08 vs 0.11), and the need to limit the volume of salt in the core results in larger inventories of fuel isotopes in the external fuel stream which are not compensated entirely by lower inventories in the fertile stream (compare Tables 6.1.2 and 6.2.2). Thus the over-all specific power for the salt system (2.7) is considerably less than that of the aqueous reactor (4.0).

LBBR. - The neutron yield, $\bar{\eta}\epsilon$, might conceivably be increased by 0.01 by further optimization of the moderator-fuel ratio without increasing losses to the fuel carrier (bismuth). Doubling the concentration of ThO_2 in the fertile-stream slurry would decrease losses to the fertile-stream carrier perhaps by 0.01. If, as postulated by Thomas, et al. (22), a substantial fraction of the fission fragments, especially iodine and xenon, escape from the particles of ThO_2 , these might be removed by some as yet unspecified process and so reduce the poison fraction by 0.004. The breeding gain thus might be as high as 0.075 instead of 0.051. The corresponding fuel yield would be 4.1%/yr instead of 2.8%, if there were no increase in inventory of U-233 in the blanket or elsewhere.

Further improvement in the fuel yield depends upon reduction in the inventory of fuel. The only item that appears amenable is the fertile-stream inventory, and a reduction is contingent upon the invention of some processing scheme more economical than Thorex which would permit more rapid processing.

The nuclear deficiencies of the LBBR vs the MSBR stem partly from the inferiority of bismuth as a heat transfer agent compared to salt (Sec. 3.4) and partly from the inferiority of ThO_2 as a fertile material compared to

ThF₄. Bismuth has a low volumetric heat capacity, and large volumes must be provided both in the core and in the external fuel circuit in order to hold the velocity within bounds. The bismuth in the core results in nuclear losses as large as those experienced with salt, and the external volume results in large inventories of fuel. Use of ThO₂ also ties up a large inventory of U-233 in the fertile stream compared to that of the MSBR, where the U-233 is stripped rapidly and economically by means of the fluoride volatility process.

GGBR. - An examination of the breakdown of the nuclear losses does not disclose any that might be reduced substantially except perhaps that due to xenon in the fuel plates. If a porous fuel matrix could be developed such that xenon, or its precursor, iodine, could diffuse rapidly to the surface and be carried away and diluted by the coolant gas, then perhaps as much as 0.04 could be added to the breeding gain - a substantial increase. It is not possible to estimate the feasibility or economics of such a development at this time, but such an improvement would make the GGBR competitive with the MSBR in regard to nuclear capability.

More rapid processing of the fuel to reduce losses to other fission products (0.02 to 0.04) is of little avail since losses in the processing plant (about 0.2% per pass) increase rapidly with the number of passes (Cases 1 through 4, Table 6.4.1). Also, the inventory of fuel in the processing plant is proportional to the processing rate, and is substantial.

As a result of these, and other effects, the fuel yield passes through a maximum as the fuel-stream processing cycle time decreases.

The thermal specific power is of the order of 2.7 Mwt/kg when fuel yields are high. The largest inventories of fuel are those in the fertile stream (over 500 kg) and in the fuel stream processing (over 400 kg), and together these compose somewhat over half of the total (about 1600 kg). Little can be done to decrease the fertile stream inventory except by increasing the Thorex processing rate, and the cost curve is very steep in this range. The inventory in the fuel processing system is largely due to the 90-day hold-up of spent fuel before it is sent to the Thorex plant. The organic solvents used there must be protected from radiation damage, since degraded organic material carrying uranium passes to the waste, and

losses become large. Improvement in specific power thus depends upon increasing the capacity and reducing the cost of the Thorex process, or finding a substitute process that accomplishes the same ends.

In relation to the graphite-moderated, fluid-fuel systems (MSBR and LBBR) the GGBR suffers from the disadvantage that nearly the full loss of neutrons to xenon is incurred, and losses to other fission products are substantial, even at low burnups, compared to the losses in the fluid fuel reactors where the fuel is processed continuously at a high rate and where the fission products are diluted by the volume of the fuel stream in the external heat-removal system. This disadvantage is offset by the advantage that part of the moderator (graphite) can be used as fuel-carrier and that the coolant (helium) absorbs a negligible fraction of neutrons. Thus the GGBR would have nuclear capability nearly as great as that of the MSBR were it not for the fact that fuel inventories in the fertile stream in the reactor and in fuel stream processing are higher for reasons discussed above.

DGBR. - As in the GGBR, losses to xenon and other fission products are substantial. The loss of neutrons to protactinium was about 0.005 less than in the corresponding GGBR calculation, but this lower loss was achieved by a more rapid rate of processing the fertile stream, and the cost was substantially higher. At the same cost, the loss would be about the same. On the other hand, it was possible to achieve slightly better values of η for U-233 in the DGBR by virtue of the superior moderating power of heavy water. This resulted in the accumulation of relatively smaller amounts of U-235 and U-236 than in the GGBR and a somewhat better breeding gain. This advantage was sharply limited, however, by the concomitant presence of zirconium, both in the fuel and fertile stream channels. In this concept, the combination of heavy water and zirconium is somewhat better, but not substantially better as a moderator, than graphite.

The thermal specific power of 1.7 Mwt/kg of fuel is inferior to that of the GGBR, and this results from the necessity of restricting the volume of graphite fuel-carrier in the DGBR. Thermal stress and radiation damage will be greater in the DGBR, and may limit the specific power to lower values.

The inventory of fuel in the core region of the DGBR is approximately five times larger than that of the GGBR. As explained in Sec. 6.5.2, this

results from a number of effects, but mainly from the necessity of having an appreciable loading of fertile material in the core region to compete with Zircaloy structure for neutrons. Other large items of inventory possibly amenable to reduction are those in the fuel stream processing (mostly in the 90-day hold-up for radioactive cooling), and in the fertile stream in the reactor. Not much can be done to reduce these at costs below 1.5 mills/kwhr except, as mentioned in Sec. 7.1.1 above, by reducing the cost and increasing the efficiency of the Thorex process, or else by finding a better process.

7.2 Economic Potential

With the exception of the MSBR, all of the concepts studied exhibited fuel costs that appeared to be approaching minimum values in the domain of positive fuel yields, as shown in Fig. 1.1. Qualitatively, this happens because, as fuel processing rates are slowed and thorium inventories are decreased, further savings that can be effected become smaller, while on the other hand loss of breeding credit increases even more rapidly.

At the lower fuel yields, the amount of power generated in the fertile stream becomes so large (up to 35% of the total) that the assumptions on which the power densities in the external circuits were estimated by Spiewak and Parsly (11) are no longer valid. Similarly, in the gas-cooled reactors the analysis of the heat removal system was based upon the assumption that ~~only a small proportion of the total heat (~10%) would be generated in the fertile streams.~~ Systems generating 35% in the blankets would require special thermal analysis and probably new conceptual design. For these reasons the various curves in the figures cited were terminated in the lower ranges of fuel yield, and the termini were regarded as minimum attainable fuel cycle costs.

In the MSBR, where part of the fuel cycle cost is contributed by discard of salt, the fuel cycle cost was still falling as the fuel yield approached zero. It lay outside the scope of this study of extend the range of fuel yields below zero.

7.2.1 Comparison of Systems

The minimum estimated fuel cycle costs together with the cost corresponding to a fuel yield of 4%/yr are listed for several concepts in Table 7.2.1.

Table 7.2.1 Minimum Fuel Cycle Cost for Five Thermal Breeder Reactors

Concept:	AHBR	MSBR	LBRR	GGBR	DGBR
	<u>Minimum Calculated Fuel Cost,</u> <u>mills/kwhr</u>				
Uranium Usage	0.12	0.07	0.12	0.08	0.15
Thorium Usage	0.07	0.07	0.03	0.05	0.06
Carrier Usage	0.25	0.10	0.08	0.00	0.18
Thorium Replacement	0.01	0.01	0.01	0.01	0.01
Carrier Replacement	0.06	0.15	0.01	0.06	0.01
Fuel Stream Processing	0.60	0.27	0.10	0.33	0.38
Fertile Stream Processing	0.01	0.14	0.77	0.75	0.67
Breeding Credit	0.20	0.05	0.04	0.03	0.13
Minimum Fuel Cycle Cost	0.92	0.76	1.08	1.25	1.33
Fuel Yield, %/yr	7	2.7	1.4	1.9	3.5
	<u>Estimated Fuel Cycle Cost at</u> <u>Fuel Yield of 4%/yr</u>				
Fuel Cycle Cost	0.9*	0.9	1.5	1.5	1.3

* Minimum Fuel Yield = 7%/yr

It is seen that, when minimum cost systems are compared, only the AHBR qualifies as a breeder having a doubling time of 25 years or less. However, if this requirement is waived, the MSBR appears to have a fuel cost advantage of about 0.3 mill/kwhr over the AHBR.

If the doubling time requirement is not waived and the concepts are compared at a fuel yield of $\geq 4\%/yr$, then the AHBR and the MSBR have an advantage of 0.5 mill/kwhr or more in fuel cycle cost over the other concepts.* Moreover, AHBR has an advantage over the MSBR in that it can achieve a competitive cost of 0.9 mill/kwhr at a fuel yield of 7%/yr (doubling time of about 14 years).

7.2.2 Possible Improvements in Fuel Cycle Costs

AHBR. - Efforts to reduce the fuel cost in the AHBR should be centered on improving on the Thorex process. Some saving might be effected by designing the blanket system to use less D_2O and to reduce the replacement rate (3%/yr). Or, a saving would be realized if the cost of D_2O were to fall appreciably, as it may. Other costs appear to be rather inflexible and are determined by the physical requirements for breeding.

MSBR. - The estimate of fuel costs in the MSBR was based on a considerable extrapolation of existing technology, and is optimistic. No single item in the cost stands out as offering opportunity for further savings.

~~The reliability of the estimate, relative to that of the AHBR, depends~~ upon the validity of the unit cost and efficiency assumed for the HF dissolution process for treating the fuel of the MSBR. The cost was based on an extrapolation by Burch, et al. (17) of a study by Weinrich and Associates (31) of a smaller plant, and is thought to be reasonably well established, though of course not as firmly as the costs for the Thorex process. The efficiency is much less certain, and the assumed values were derived from predictions based on theoretical inorganic chemistry, supplemented by a few laboratory experiments (32). Bench-scale tests are lacking.

Compared to the AHBR, the minimum-cost MSBR is thus merely a self-sustained reactor, and is not suitable for use in an expanding nuclear system where new reactors are inventoried with excess fuel produced in reactors of the same type.

* See Sec. 7.2.3

At a fuel yield of 4% (doubling time of 25 years), the fuel cost of the MSBR about equals that of the minimum cost AHBR (0.9 mill/kwhr). But, as mentioned above, the reliability of this estimate of performance is greater for the AHBR than the MSBR. The cost curve for the MSBR rises steeply at 4%, while that of the AHBR does not, and errors in estimating processing efficiencies will affect the MSBR results more strongly than the AHBR.

LBBR. - The single largest item of fuel cost was that for processing the fertile stream (Thorex), and amounted to 0.7 mill/kwhr with about 20% of total heat being removed from the fertile stream. This cost could perhaps be reduced to 0.1 mill/kwhr by going to higher blanket loadings and allowing up to 35% of the reactor heat to be generated in the fertile stream, but a new conceptual design would probably be required.

In this study, the fuel stream processing cost of 0.1 mill/kwhr was adopted from the work of Thomas, et al. (22). This compares unfavorably with the corresponding cost in the AHBR (less than 0.01 mill), though appreciably better than that of the MSBR (about 0.25 mill), but the estimate appears to be less firm than either of these for the reason that experimental verification and cost analysis are not presently available in comparable depth and scope.

It was not possible to make the fuel-stream processing rate a parameter in this study because the poison fraction was given by Thomas only at the rate corresponding to the cited cost. However, since at the most only 0.1 mill could be saved by decreasing the fuel-stream processing rate to zero, and since such increase would not only wipe out the breeding credit (0.04 mill), but would also make the system dependent on outside sources of fuel, it seems clear that lower costs cannot be achieved by variation of the fuel processing rate.

Mainly because bismuth is a poor heat transfer agent, and large volumes are required in the external fuel system, inventory charges for uranium are higher in the LBBR than in the MSBR by about 0.05 mill/kwhr. This, coupled with a low breeding credit results in a minimum calculated fuel cycle cost of about 1.1 mills/kwhr. By making allowance for errors and uncertainties in the data, and for further possible optimization of the moderator-uranium ratio, etc., a minimum fuel cycle cost of about

1.0 mill/kwhr at a positive fuel yield of, say 1%/yr was estimated. But this minimum-cost LBBR is not suitable for use in self-expanding breeder economy.

GGBR. - Further efforts to reduce the fuel cost must be centered on items lying outside the scope of the present study, namely, on ways to improve or replace the Thorex process and ways to increase the breeding credit by effecting the escape of xenon and iodine from the fuel matrix into the coolant.

DGBR. - Efforts to reduce the cost further should be centered on improving or replacing Thorex and consideration of the fuel processing scheme to eliminate the 90-day hold-up, if possible.

7.2.3 Effect of Increases in Nuclear Source Material Costs

Although the future trend in the cost of nuclear source materials is not known since this depends on (1) the rate of growth of the nuclear industry, (2) the neutron economy of the reactors comprising the industry, (3) the consumption of nuclear fuels for all uses, and (4) the availability of nuclear materials as a function of price, at some time in the future the primary source of nuclear fuels will be very low grade ores from which uranium might be extracted at about \$120-\$150/lb of U_3O_8 . At this time the cost, or value, of enriched fuel would be about four times present costs or about \$60/g.

The effect on the fuel costs, in the five thermal breeder reactors studied, of increasing the value of fuel isotopes ($U-233$ and $U-235$) from ~~\$15~~ to \$60/g is shown in Fig. 7.2.1. Since the use charge for fissionable material was taken at 4%, increasing the value of fuel lowers the fuel cost when the fuel yield is greater than 4%, and increases it when it is less.

It is seen that, in all the systems except the AHBR, the minimum cost increases by 0.1 to 0.2 mill/kwhr. The minimum cost in the AHBR decreases from 0.9 to about 0.5 mill/kwhr, and the optimum fuel yield increases from 7% to 12%/yr. Thus, only the AHBR is benefited by an increase in fuel value. Also, the AHBR is the only system capable of achieving fuel yields such that the fuel cost would be independent of fuel value when the use charges are as high as 12.7%/yr.

7.3 Development Status

Development is here defined as the work required to bring the design art for a particular concept to the stage where a full-scal prototype

Fig. 7.1.2. Thermal Breeder Evaluation

— Fuel at \$15/gram, 4% interest

---- Fuel at \$60/gram, 4% interest

station can be designed. In order to proceed, it must be assumed that all of the major problems are known. The probability of this assumption is roughly proportional to the amount of development effort that has been expended to date.

It was felt that the successful demonstration of a reactor experiment is a necessary step in the development of any concept. This experiment should incorporate every essential, untested feature of the reactor and its associated equipment (heat transfer, fuel handling, etc., but not including fuel processing).

Similarly, in the development of processes for purifying the fuel and fertile streams, the successful operation of a pilot plant is a necessary step.

By considering the number and nature of these problems in each concept, an estimate of relative rank in development status was derived.

The development of the AHBR is well advanced, and, with the exception of the design of the core-vessel, the remaining problems will not require large effort. The core-vessel design problem is currently (February 1961) under study at ORNL.

~~In the light of present evaluations, the problems facing the proponents~~
of the salt reactors do not appear to be severe, but there are more of them. The most difficult appears to be that of developing a suitable graphite-INOR joint. Work in progress should shortly clarify the position of the MSBR relative to that of the AHBR.

The other systems appear to be considerably farther from fruition. Each has many problems, and it may well be that some of these are more intractable than they now appear to be. It is difficult to rank the GGBR relative to the LBBR, but it does appear that the DGBR has, in addition to all the problems encountered in the GGBR, the further problem of providing compatibility between a high temperature fuel and a low temperature moderator.

The number of problems in a given reactor concept is a rough, relative measure of the amount of development work needed for that concept. However, different weights should be attached to the various problems. For example, the engineering design required for the fuel and fertile stream processing in the AHBR is considerably less than that required for any of the others.

The major development problems were discussed briefly in Sec. 6. The crucial problems for each concept are listed below:

Crucial Development Problems

AHBR Integrity of the core vessel.

MSBR Integrity of core structure; HF dissolution process for purification of fuel salt; control of neutron losses to xenon.

LBBR Fuel- and fertile-stream processing; control of neutron losses to xenon; all-tantalum heat exchanger; integrity of core structure.

GGBR Fuel handling machinery; fuel-stream processing and remote fabrication of fuel elements; integrity of core structure.

DGBR Fuel handling machinery; fuel-stream processing and remote fabrication of fragile fuel elements; integrity and safety of core structure; radiation damage and thermal stress in fuel.

7.4 Summary of Comparisons and Conclusions

In Table 7.4.1 are listed the ranks of superiority of the five reactor concepts studied herein in the three categories of comparison.

Table 7.4.1 Rank of Superiority

Concept:	AHBR	MSBR	LBBR	GGBR	DGBR
Nuclear Capability	1	2	4	4	3
Economic Potential*	1	1	3	3	2
Development Status**	1	2	3	3	4
Over-All Rank	1	2	3	3	4

* Based on fuel cycle costs in systems having fuel yields of 4%/yr or more.

**Based on relative effort required to complete development.

The aqueous homogeneous breeder reactor (AHBR) clearly exhibits a superior nuclear capability, with fuel yields ranging from 7% to 16%/yr (corresponding to doubling times of 14 to 6 years), at costs ranging from 0.9 to 1.5 mills/kwhr. These estimates are based on considerable experimental data from the HRE-2 and relatively firm cost figures from a Thorex pilot plant. Even when allowances are made for uncertainties in the nuclear data and errors in costing, it can be asserted with confidence that the AHBR is capable of producing 10%/yr of excess fuel at a fuel cycle cost of not more than 1.5 mills/kwhr, and possibly as low as 1.0 mill/kwhr. Of this cost, about 0.6 mill/kwhr is contributed by the Thorex processing. Improvement in efficiency in the Thorex plant would effect a substantial reduction in the AHBR fuel cost - possibly to 0.7 mill/kwhr.

The molten salt breeder reactor (MSBR) appears to be capable of fuel cycle costs down to 0.6 mill/kwhr if the fuel yield is allowed to approach zero. This estimate is based on reasonably reliable data and cost bases, but these are less firm than with the AHBR, and the nuclear performance of the MSBR, with 30% epithermal fissions, is more sensitive to errors in the nuclear data (especially U-233 cross sections) than is the AHBR with only 5%-15% epithermal fissions.

~~If attention is restricted to thorium fueled reactors suitable for use~~
in self-expanding nuclear power economy (defined arbitrarily as those capable of producing 4% excess fuel per year or more), then the minimum cost in the MSBR is about 0.9 mill/kwhr and is not less than the minimum cost in the

AHBR. The AHBR has an advantage over the MSBR in that, at comparable minimum costs in an expanding economy, the AHBR can achieve a higher growth rate (7%/yr) than can the MSBR (4%/yr). Further, with modest increases in cost, the AHBR can achieve growth rates up to 16%/yr, whereas the MSBR, with a steeper cost curve, is limited to a growth rate of 7%/yr. Also, more effort would be required to bring the development of the MSBR to the point where a full-scale prototype power plant could be designed than would be required for the AHBR.

The AHBR is clearly the concept preferred for development of thermal breeder reactors, based on the criteria used here. The fuel cycle cost is, however, an incomplete criterion, and was used only because estimation of net power costs did not lie within the scope of the present investigation.

Should the capital and operating costs of the AHBR reactor and steam generating plants substantially exceed those of the MSBR, the above-stated preference for the AHBR might be revised in favor of the MSBR.

7.5 Comparison with Natural-Uranium Heavy-Water Power Reactors

In respect to economic power, the thermal breeder reactors must compete with fast breeder reactors on the one hand, and with converter reactors on the other. A prominent example of a system in the latter group is the Canadian-deuterium-uranium (CANDU) reactor proposed by W. B. Lewis, et al as reported in the Oct., 1960 issue of Nucleonics.

The CANDU approach to competitive nuclear power is well adapted to the Canadian financial and geographical milieu. Advantage is taken of a financing and tax structure that permits a low annual fixed charge averaging about 8% on capital investment, labor costs that are 40% lower than in the U.S., a dollar exchange rate that gives Canada a 3% advantage over the U.S., a procurement policy which permits purchase of equipment from European as well as Canadian vendors at an estimated saving of 10% over U.S. costs, the availability of abundant supplies of natural uranium at a cost about 25% less than the official U.S. A.E.C. price, and a cooling-water temperature that permits a condenser design pressure of 1 in. Hg absolute, compared to 1-1/2 in. Hg or greater in the U.S., which gives an advantage of about one per cent in thermodynamic efficiency in the reactor steam plants to the Canadians.

Because of these differences, the power costs estimated for CANDU by the Canadians cannot be compared directly to the cost of power in U.S. reactor systems. Accordingly, a reevaluation, based on published information, of the CANDU system was undertaken. This included not only a translation to U.S. cost bases but also some reoptimization of the nuclear and thermal design with respect to costs. Details of this study are given in Appendix J.

7.5.1 Description

The CANDU reactor is a full-scale, 200 Mwe nuclear power reactor using Zircaloy clad, natural-uranium dioxide fuel. It is moderated with low-temperature D_2O , and cooled with D_2O in pressure tubes (152). CANDU is a scale-up of a 20 Mwe prototype reactor, the NPD-2, now under construction at Rolphton, Ontario. The NPD is scheduled for full-power operation by mid-1961 (153),

and will be used to test the fuel charging machines proposed for the first CANDU plant.

The moderator, D_2O at $80^\circ C$, is contained in a calandria of Zircaloy about 20 ft in diameter and 16-1/2 ft long. Some 200 Zircaloy pressure tubes 3.25 in. I.D. with walls 0.16 in. thick pass through thin-walled calandria tubes arranged in a square lattice on a pitch of 9 in. Fuel elements are charged into the pressure tubes and are cooled by heavy water flowing at 20-30- fps. The pressure in the tubes is about 1500 psi, the inlet temperature is $480^\circ F$, the outlet temperature is $570^\circ F$. The elements travel in opposite directions in adjacent channels. Charging is done semi-continuously at full power. The transit time in each channel is adjusted so that all fuel elements will be subjected to the maximum allowable exposure. The elements consist of bundles of Zircaloy tubes containing pellets of natural uranium oxide (UO_2). Bundles containing 7 tubes 0.95 in. I.D. by 19-1/2 in. long and 19 tubes 0.57 in. I.D. by 19-1/2 in. long have been proposed. The cladding thickness is 0.015 to 0.025 in.

After exposure the fuel elements are discarded (stored), and no attempt is made to recover either plutonium or uranium from them.

Steam is produced in the boilers at 580 psi and $482^\circ F$. The thermodynamic efficiency of the steam plant is 33-1/2%, and the station net efficiency is 29.1%.

For the NPD and CANDU reactors using fuel rods 0.6 in. in diameter, Mooridian and Robertson (154) have estimated the following costs.

Projected Canadian Fuel Costs (154)

Item	\$ / lb UO_2	
	NPD (1960)	CANDU (1964)
Raw material		
Yellow cake	9.00	6.00
Conversion to powder	3.00	1.50
Sintering, grinding	4.00	3.00
Zircaloy components	12.50	7.50
Assembly, inspection	<u>9.00</u>	<u>9.00</u>
Total	37.50	27.00

In computing CANDU costs, Lewis (152) takes a figure of \$30/lb UO_2 for the fabricated fuel, a burnup of 9,750 Mwd/tonne, a station efficiency of 29.1%, and no credit for value of spent fuel. These numbers lead to an estimated Canadian cost of 1.1 mills/kwhr, exclusive of inventory and replacement charges for D_2O , and exclusive of interest on the fuel inventory. D_2O replacement was estimated by Lewis at 2% of the inventory per year, or 0.17 mill/kwhr. The value of the D_2O inventory (180 tonnes at \$28/lb) is about \$10 million, which, with the interest rate of 8%/yr used by Lewis, contributes about 0.55 mill/kwhr to the power cost. Inventory charges on fabricated fuel in the reactor at 8%/yr add another 0.2 mill/kwhr to the fuel cycle cost. The total of those items customarily included in fuel cycle costs in this country amounts to 2 mills/kwhr on Canadian cost bases.

But these costs, as pointed out above, are not valid for reactors to be built and operated in the United States.

7.5.2 U.S. Design and Cost Bases

The CANDU concept has been reevaluated using, where applicable, the bases used to evaluate the five thermal breeder reactors considered in previous sections of this report. Other bases, applicable only to CANDU, were modified in accordance with U.S. economics and practice.

A station power of 1000 Mwe was selected. Although Lewis's estimate of 29.1% for the station net efficiency was felt to be optimistic for locations in the U.S., this number was adopted. Cores capable of delivering 250 Mwe (383 fuel channels per core) and 333 Mwe (505 channels) were evaluated with respect to attainable exposure at the same specific heating rate (15 kw/ft of fuel rod -- maximum) forecast for CANDU-1 (154). Exposure limits in CANDU cores having D_2O reflectors 60, 68, and 75 cm thick were determined by means of the Expire program (155) for the IBM-704. The thickness was optimized with respect to the fuel cycle cost.

The system was also reoptimized with respect to fuel rod diameter. Rods of 1.0, 0.82, and 0.6 in. O.D. were selected for study.

In regard to the inventory of D_2O , the volume of D_2O required to fill the external heat transfer system was estimated using the power density capability predicted by Spiewak and Parsly (11) for the AHBR. This capability is

somewhat more optimistic than that used by the Canadians, and resulted in lower inventories and a saving of 0.03 mills/kwhr in fuel cycle costs.

In the Expire calculations, the concentration of H_2O in the heavy water was taken as 0.14%, compared to 0.25% used in the AHBR and DGBR concepts in previous sections. The same value, however, of \$28/lb was assigned to this material. Also, 20% less Zircaloy than specified by Lewis, et al was inadvertently allotted in the nuclear calculations.

The cost bases selected correspond to quantity production at current prices. The recently announced official U.S. A.E.C. purchase price (156) of \$8/lb of U_3O_8 (yellow cake) was adopted.

As shown in Appendix J, the cost of fabrication of Zircaloy clad fuel rods, when expressed in \$/ft of rod, is proportional to the diameter of the rods. Thus, the fabrication cost, in terms of \$/lb of UO_2 , becomes approximately inversely proportional to the diameter. The formula

$$\text{Fabrication cost, } \$/\text{lb } UO_2 = 20/D$$

was adopted, where D is the pellet diameter in inches. Thus, for pellets 0.6 in. O.D. x 0.015 in. wall thickness, the estimated fabrication cost is \$35/lb of UO_2 , which conforms to the lower limit of estimates in the AEC Nuclear Power Plant Cost Evaluation Handbook (157), of fabrication of the Zircaloy sheath, and assembly, inspection and testing of the fuel rod clusters.

Taken with the cited cost for U_3O_8 , this leads to a total cost of \$42.70/lb of UO_2 for fabricated fuel elements (0.6 in. O.D.). This is to be compared to the quotation of \$37.50 for current costs of NPD fuel elements cited by Mooridian and Robertson (154), which were based on a cost of yellow cake of \$8.65/lb and to the estimate of \$30/lb used by Lewis and cited above. Thus the fabrication cost adopted in the reevaluation is \$6.50 more per lb of UO_2 than the current Canadian estimate, and reflects in part, at least, effect of higher costs in the U.S.

Inventory charges of 12.7%/yr on natural uranium and heavy water were taken rather than 8%/yr used by the Canadians. In this connection, natural uranium was taken as the counterpart of thorium in the thermal

breeder reactors. There is thus no material in CANDU corresponding to the enriched fuel (U-233) used in the breeders, an advantage for the CANDU system. Also, following Lewis's procedure, no value was assigned to the spent fuel,^d even though it contains appreciable amounts of Pu-239 in addition to unconverted U-238. On the other hand, no charge was made for storing the spent fuel. In the thermal breeders, of course, there is an appreciable credit accruing from the production of excess fuel.

The replacement rate for heavy water was optimistically taken at 2% of the total inventory per year, as allowed by Lewis (op. cit.), compared to a replacement rate of 3%/yr in the AHBR and DGBR cost estimates.

These bases, together with miscellaneous items are summarized in Table 7.5.1. The corresponding Canadian bases are given for comparison

Table 7.5.1.
Cost Bases for TBREP Reevaluation of CANDU

	U.S. Bases ^a	Canadian Bases ^b
Unit Costs		
Natural uranium, \$/lb U ₃ O ₈	8.00	8.65
D ₂ O (0.14% H ₂ O), \$/lb	28.00	28.00
Fuel fabrication ^c , \$/lb UO ₂	20/D	12.5/D
Annual Charges, %/yr		
Fabricated fuel or fertile material ^d	12.7	8
Heavy water	12.7	8
Heavy water replacement	2.0	2.0
Plant Utilization Factor	0.8	0.8
Station Net Efficiency	29.1	29.1

a. Current quantity costs

b. Based on Lewis's estimate of \$30/lb UO₂ for fabricated fuel, presumably ~ 1964

c. D = pellet diameter in inches

d. Annual charge 12.7% of 1/2 of value of fabricated fuel in CANDU reactors, since spent fuel is discarded and has no value.

The major differences between the bases adopted and those used by the Canadians consist in about 50% higher fuel fabrication costs and interest charges in the U.S. bases.

7.5.3 Results and Conclusions

Details of the results for the various cases are given in Appendix J. In general, the nuclear calculations by means of Expire predicted reactivity lifetimes about 10% less than those predicted by the Canadians for comparable cases. The source of this discrepancy is not presently known, but may reside in the method of estimating effective one-group cross sections for U-235 and U-238, or fission products. In any event, the discrepancy is not important.

Results for the case giving the least fuel cycle cost are summarized in Table 7.5.2. This case has been partially optimized with respect to reactor power, diameter of fuel rods, heavy water inventory, and reactivity lifetime. It is thought to exhibit near minimum fuel costs for the concept. For comparison, items from Canadian estimates on CANDU-1 are also given.

Table 7.5.2.
Characteristics of CANDU Concept Optimized with
Respect to U.S. Cost Bases

	U.S. Bases	Canadian Bases
Station power, Mwe	1000	203
Number of reactors per station	5	1
Net station efficiency	29.1	29.1
Thermal power per reactor, Mwt	1150	698
Reactor core		
Tank diameter, ft	23	19.7
Active diameter, ft	19.0	14.8
Active length, ft	16.4	16.4
Lattice pitch, in.	9	9
Tank volume, ft ³	6790	4980
Number of fuel channels	505	306
Reflector thickness, in.	24	30

	U. S. Bases	Canadian Bases
Fuel		
Fuel rod diameter, in. O.D.	0.6	0.6
Rods per cluster	19	19
Fuel rod length, in.	19-1/2	19-1/2
UO ₂ in active core, tonnes	77.9	47.2
UO ₂ in reactor, tonnes	93.4	56.6
Estimated burnup, Mwd/TU	10,100	9,750
Heavy Water		
D ₂ O in active core (as moderator)	120.2	72.2
D ₂ O in reflector, tonnes	59.4	67.2
D ₂ O in heat transfer system, etc., tonnes	81.4	40.6
Total D ₂ O, tonnes	261.0	180

These inventories and burnup lead to a fuel cycle cost of $\frac{1.99}{2.01}$ mills/kwhr when computed on Canadian bases, and a cost of $\frac{2.74}{2.99}$ mills/kwhr when computed on U. S. bases. Breakdowns of these costs are given in Table 7.5.3. Also given for comparison is a breakdown, in comparable categories, of the fuel cycle cost for the AHBR.

It is seen that the AHBR enjoys a fuel cost advantage of 2 mills/kwhr over the CANDU system in the U.S., and 1 mill/kwhr advantage when compared to a CANDU reactor built, financed, and operated in Canada. If the AHBR costs were translated to Canadian bases, the sum would probably be less than 0.7 mills/kwhr. It is concluded that the Canadians approach is not appropriate to the situation in the U.S. and cannot compete with the breeder reactors studied in respect to fuel cycle costs.

Table 7.5.3
Fuel Cycle Costs for CANDU
and AHBR Systems

Cost Bases	CANDU Canada	CANDU United States	AHBR United States
	mills/kwhr		
Inventory charges			
Enriched fuel (at 4%/yr)	-	-	0.12
Thorium or natural uranium ^a	0.10	0.25	0.07
D ₂)	0.63	0.87	0.26
Replacement charges (raw material)			
Th ^b or natural uranium	0.32	0.28	0.01
D ₂ O ^c	0.17	0.14 ^e	0.06
Fuel fabrication or reprocessing	0.75	1.20	0.61
Credit for excess fuel ^d	-	-	- 0.20
Net fuel cycle cost, mills/kwhr	1.99	2.74	0.93

a. Based on 1/2 value of fabricated material, see Table 7.5.1

b. At \$12.25/lb Th as pellets of ThO₂

c. At 3%/yr for AHBR, 2%/yr for CANDU

d. A \$15/gram for AHBR product

e. Thinner reflector

8. A C K N O W L E D G M E N T S

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